Preliminary

CERTIFICATE HOLDER: United States Enrichment Corporation
Paducah Gaseous Diffusion Plant
Paducah, Kentucky

SUBJECT: COMPLIANCE EVALUATION REPORT: APPLICATION DATED OCTOBER 20, 2000; HIGHER ASSAY UPGRADE PROJECT

EXECUTIVE SUMMARY:

This report documents the United States Nuclear Regulatory Commission (NRC) staff compliance evaluation of the U.S. Enrichment Corporation (USEC) certification amendment application for the Paducah Gaseous Diffusion Plant located in Paducah, Kentucky. Currently, the Paducah Gaseous Diffusion Plant in Paducah Kentucky, enriches uranium up to a maximum of 2.75 percent by weight (wt%) ^{235}U and ships this material to the Portsmouth Gaseous Diffusion Plant located in Portsmouth, Ohio, to continue to be enriched up to 5.0 wt%. USEC recently decided to stop operations at the Portsmouth facility, and increase the enrichment capabilities of the Paducah facility. USEC has submitted to NRC a certification amendment request dated October 20, 2000, asking to increase Paducah's assay limit from 2.75 wt% to 5.5 wt%.

USEC requested a very aggressive schedule for the review of this amendment request because of their desire to demonstrate that the Paducah facility was capable of producing the higher enrichment prior to taking the Portsmouth facility offline during the spring of 2001. Therefore, to perform an efficient and effective review, the NRC staff began reviewing relevant nuclear criticality safety evaluations and approvals (NCSE/As) prior to receiving the amendment application. The staff reviewed 53 of the NCSEs, performed on-site reviews, developed Requests for Additional Information (RAIs), reviewed the RAI responses, requested changes to the NCSE/As, and reviewed the revised NCSEs.

Based on this review, the staff concludes that the Paducah facility's assay limit can be safely increased to 5.5 wt%. However, in the process of conducting its review, the staff did note several programmatic deficiencies with USEC's nuclear criticality program. USEC submitted sufficient commitments to resolve these issues to ensure continued safe operation of the Paducah Gaseous Diffusion Plant at an assay limit of 5.5 wt%.

BACKGROUND:

On February 10, 2000, the NRC denied USEC's first certificate amendment request for the Paducah Higher Assay Upgrade Project (HAUP). This amendment request, dated January 21, 2000, requested the deletion of Technical Safety Requirements (TSR)s 2.3.4.6 and 2.4.4.3. These TSRs established facility and process- specific enrichment limits less than the facility wide limit. On March 6, 2000 NRC and USEC management met to discuss this amendment request.

On May 19, 2000, NRC held another public meeting with USEC to discuss the proposed schedule for submission of the revised HAUP amendment request. In this meeting USEC stated that it was their intention to request an amendment that allowed the Paducah facility to increase its enrichment at the facility from 2.75 wt% to 5.5 wt% and committed to submitting the revised amendment request to NRC by October 2000. Because of the aggressive schedule needed to complete the review, the NRC agreed to begin review of relevant nuclear criticality safety evaluations and approvals (NCSE/A) in preparation for receiving the amendment request in October. By letter dated June 13, 2000, USEC provided NRC with a list of all of the NCSE/As that would be impacted by the amendment request and provided a proposed schedule for the submittal of these documents to NRC. In a letter dated July 17, 2000 the NRC stated that in order to effectively and efficiently review the NCSE/As these documents need to be submitted to the NRC to be placed on the docket.

USEC submitted completed NCSEs to NRC for review in letters dated August 4, August 29, October 3, October 10, November 7, and December 15, 2000, and January 4, January 8, January 10, and January 25, 2001. This documents were submitted as proprietary information. USEC also submitted non-proprietary nuclear criticality safety approvals for review in letters dated August 4, October 3, October 10, November 7, and December 15, 2000 and January 4, January 10, and January 25, 2001.

The staff requested additional information on the amendment request and NCSE/As by letters dated September 14, October 6, November, 1, November 21, December 1, December 6, December 12, and December 21, 2000. USEC responded by letters dated September 29, October 27, November 22, and December 13, 2000, and January 4, 2001. In addition to reviewing the documents, the relevant staff also visited the site over the weeks of September 5, September 25, October 23, and November 27, 2000.

Also to ensure that the review went smoothly and that issues were elevated and resolved quickly, there were periodic management meeting between NRC and USEC management on October 3, October 16, and November 20, 2000 and January 29, 2001.

DISCUSSION:

The review of this amendment request required the staff to review the Certification Amendment Request (CAR) which contained proposed revised pages to the safety analysis report (SAR), revisions to technical safety requirements (TSRs), revisions to the fundamental nuclear material control plan, and minor changes to the physical protection plan, and the security plan. The staff also reviewed specific NCSE/A which comprised the safety basis for this action. The following discussion includes a description of the review which was performed in each of these areas.

1. <u>EVALUATION OF SAR CHANGES</u>

1.1 Changes to SAR Chapter 1, “Introduction and General Description of the Facility”

Most of the SAR Chapter 1 changes are merely descriptive in nature, changing the authorized amounts and types of materials that can be handled, stored, and shipped. The SAR is being modified to allow the filling and on-site storage of cylinders up to 5.5wt% ^{235}U. Although some changes to Chapter 1 refer to “storage, and shipment of cylinders containing uranium enriched up to 5.5 percent by weight ^{235}U”, USEC committed in a letter dated January 2, 2001(GDP-00-0235), to continue adhering to current assay shipping limits. Cylinders with an assay greater than 5.0 wt% ^{235}U will not be shipped or received, except small (<8 inch diameter) cylinders approved for > 5.0 wt% ^{235}U. Based on this, the changes to SAR Chapter 1 are acceptable.

1.2 Changes to SAR Chapter 3, "Facility and Process Description"

Most of the SAR Chapter 3 changes involved changing the enrichment limit for specific operations from 2.75 to 5.5 wt% ^{235}U. For most of these changes, the corresponding NCSE/A provided the technical basis for the change and was reviewed by the staff (see Appendix B). In particular, the product cylinder filling operation was changed to permit filling cylinders to an assay of 5.5 wt% ^{235}U, although cylinders with greater than 5.0 wt% assay were required to be refed into the cascade as they exceed the shipping limit. In this and other operations, a regulatory assay limit of 5.5 wt% ^{235}U was requested to provide sufficient margin to enable production at a nominal 5.0 wt% without exceeding the regulatory assay limit. USEC provided historical data (GDP-00-0235, dated January 2, 2001) demonstrating the ability of the product withdrawal operators to control product cylinder assay to well within the 0.5 wt% margin. This data was not comprehensive and USEC acknowledged that there is likely to be greater variability in the assay at 5.5 wt% than at 2.75 wt% assay. However, based on limited historical data and statements (GDP-00-0235) concerning the time required to achieve assay shifts in the cascade, the staff has no reason to doubt the facility's ability to control assay within regulatory limits. Moreover, the violation of assay limits would be reportable and there has been no history of a systemic inability to control the assay at the Portsmouth GDP.[1]

Cylinder refeed is not anticipated to have a major impact on the cascade assay gradient, due to the low feed rate (approximately 10^{-2}-10^{-3} times the interstage flow rate) and compensating change in the product withdrawal rate. TSR 2.4.4.3 requires control of the assay in the cascade to less than 5.5wt% ^{235}U and establishes a surveillance requirement to monitor the product stream assay twice per shift; the product stream assay can be controlled by adjusting the product withdrawal rate. The major mechanism leading to an increase in the product stream assay is a reduction in product withdrawal rate caused by equipment malfunction or the introduction of light gases into the cascade. USEC's experience is that changes in the assay gradient are relatively slow processes that typically require more than 8 hours to exceed the regulatory assay limit.

The risk of a criticality event resulting from slightly exceeding the assay limit of 5.5 wt% is unlikely provided moderator control is maintained. In addition, because the facility's TSR limits require that they remain below the 5.5 wt% limit any violation of the TSR would be required to be reported to NRC and USEC would be required to take appropriate actions to maintain safety.

The only materials known with assays greater than 5.5 wt% ^{235}U are standards and sources in the C-710 Laboratory, and legacy equipment discovered in DOE Material Storage Areas

[1]Because the staff has not conducted an in-depth review of the similarities and differences between operations and programs at the Portsmouth and Paducah GDPs no credit was given for Portsmouth's record during NRC evaluation of Paducah's request.

(DMSAs). The only other credible mechanism for exceeding the assay limit of 5.5 wt%, involves the mixing of these materials with materials in other areas. There are few fissile materials at assays higher than 5.5wt% ^{235}U in other areas of the facility and these are limited by possession limits in SAR Chapter 1 and tightly controlled by specific NCSE/As (GEN-24 and GEN-20).

Besides the changed references to the global facility limit of 5.5 wt% ^{235}U, the majority of other changes to SAR Chapter 3 involved the description of operations that were required to be modified to support higher assay operations. There were a significant number of changes to cascade configurations, assay levels, and throughputs, resulting in the cascade being separated into distinct “enricher” and “stripper” sections, among other changes. As stated in Letter GDP-00-0235, no equipment changes other than valving re-alignments were required. No changes to existing NCSE/As or TSRs were required to support the changes to the cascade. Therefore, the NCS implications of the changed operation of the cascade are considered minimal.

Another series of SAR changes was made to reflect the change in cascade operations from single to double contingency (NCSE/A CAS-002). The staff’s evaluation of CAS-002 is presented in Appendix B, Section B.43. USEC acknowledged in Letter GDP-00-0235 that the effort to make the cascade doubly contingent did not entirely succeed, and that the cascade remained singly contingent for hot metal reaction and fire scenarios. This resulted in the need for an additional TSR and removal of the SAR changes.

2.0 Evaluation of Technical Safety Requirements (TSRs) Changes

Proposed TSRs were submitted with the initial amendment request and also in response to NRC staff RAI’s. All TSR changes which are associated with this amendment request are discussed in this section.

Appendix A - USEC is requesting that TSR 2.1, TSR 2.2, and TSR 2.3 in Appendix A be modified to expand the scope of cylinder mass limits to include USEC-owned 30B cylinders as well as customer-owned 30B cylinders. The staff finds that the changes have no significant safety impact and are therefore approved.

TSR 2.3.4.6 - USEC is requesting that this TSR be modified to increase the assay limit of product withdrawal equipment and facilities (excluding the North Bank NaF Traps) from 2.75 to 5.5wt% ^{235}U. The North Bank NaF Traps will continue to be operated at 2.0wt% and the tails withdrawal facility at 1.0wt% ^{235}U assay. The basis for this change was evaluated by the staff in its review of NCSE/A 3974-05, “Product Withdrawal System” and found to be acceptable (see Section B.46).

TSR 2.4.4.3 - USEC is requesting that this TSR be modified to increase the assay limit of all cascade equipment (excluding the 20-MW Freezer/Sublimer, UF_6/R-114 Separation Unit, and 24-inch Alumina Traps) from 2.75 to 5.5wt% ^{235}U. The basis for this change was evaluated by the staff in its review of NCSE/A CAS-002, “Operation and Maintenance of the UF_6 Cascade” and found to be acceptable (see Section B.43).

TSR 2.4.4.4 - USEC is requesting to change the basis statement for this TSR to add the statement that the maximum hydrogen-to-uranium (H/U) ratio for deposits in a non-fluorinating environment is 4. This change does not result in a change to any criticality limit or control. USEC stated that the technical basis for this assertion was provided in NCSE/A CAS-011, "Shutdown of Cascade with & without Inventory". Staff reviewed this document and found the TSR to be acceptable (see Section B.45).

TSR 2.5.4.3 - In response to issues identified by the NRC during its review (see Section B.8) of NCSE/A GEN-010, "Removal and Handling of Contaminated Equipment from the Cascade at PGDP," TSR 2.5.4.3 was modified to require the isolation of the local sprinkler system and to cover Planned Expeditious Handling (PEH) equipment openings with pre-staged waterproof covers, in the event of inadvertent actuation. The basis for this change was evaluated by the staff and found to be acceptable.

TSR 3.12 - In response to issues identified by the NRC during its review (see Sections B.8, B.43 and B.45) of NCSE/A GEN-010, "Removal and Handling of Contaminated Equipment from the Cascade at PGDP," NCSA CAS-002, "Operation and Maintenance of the UF_6 Cascade", and NCSA CAS-011, "Shutdown of Cascade with & without Inventory", TSR 3.12 was modified to identify elements of the fire protection program that ensure the prevention of a fire large enough to initiate sprinkler activation which could become a source of moderation for the cascade. The basis for this change was evaluated by the staff and found to be acceptable.

TSR 2.5.4.5 - In response to issues identified by the NRC during its review (see Section B.18) of NCSE/A GPS-25, "Disassembly and Repair of Process G-17 Valves", TSR 2.5.4.5 was established to ensure that equipment categorized and removed as PEH are spaced 10 feet edge-to-edge from fissile or potentially fissile equipment or containers until uranium decontamination is below a safe mass. The basis for this change was evaluated by the staff and found to be acceptable.

TSR 2.4 - The remaining TSR changes pertain to the new safe mass curve in TSR 2.4, Appendix B. The existing safe mass curve was derived for an optimally moderated deposit, while the proposed safe mass curve is for a maximum H/U = 4, and therefore allows a substantially larger deposit in the cascade without being classified as Planned Expeditious Handling (PEH). A footnote to the proposed safe mass curve states that the new curve will not be used for equipment removal. USEC Letter GDP-00-0235, dated January 2, 2001, commits that the safe mass curve will be used to determine when the safe mass conditions in TSR 2.4.4.4 are exceeded. TSR 2.4.4.4 contains required actions and associated time limits for equipment in the cascade containing greater than safe mass deposits. NRC had previously accepted the claim that the maximum moderation resulting from wet air inleakage is H/U = 4; the staff did not re-evaluate the basis for this assumption. Therefore, provided liquid water is not introduced (i.e., the only credible source of moderator is wet air inleakage) it is appropriate to use an H/U = 4 in making the safe mass determination.

Conditions A and B in TSR 2.4.4.4 establish required actions for deposits in a fluorinating environment and a non-fluorinating environment, where the hazard is moderation from wet air inleakage. However, Condition C applies to deposits in a non-fluorinating environment when the R-114 coolant pressure is not greater than the Recirculating Cooling Water (RCW) pressure. There are specific time limits established for either increasing the R-114 pressure, or draining the RCW. The requirement that the R-114 pressure is greater than the RCW pressure is a barrier against the introduction of liquid water, which can lead to deliquescence of the deposit and moderation conditions greater than an H/U = 4. Therefore, basing the determination of safe mass on the proposed safe mass curve could result in a criticality hazard due to the controls in Condition C not being applied for deposits with masses in the intermediate range. The staff considers it inappropriate to use the H/U = 4 safe mass curve to determine whether to apply controls established to prevent moderation from other than wet air inleakage.

Notes at the bottom of the graph in TSR 2.4, Appendix B, indicate that the H/U = 4 curve was computed assuming cylindrical geometry, interaction with a 5.5-gallon drum at 1500 gUO_2F_2/l (in optimal range), and mixed concrete-water reflection. DAC-832-ZA1280-006, “Subcritical and Safe Mass Curves for PGDP,” contains the technical basis for the calculated curve. USEC used the SCALE-4.4 code package to search for the maximum subcritical dimensions (corresponding to $k_{eff} + 2\sigma = 0.9634$) for a UO_2F_2-water sphere at H/U = 4 surrounded by twelve inches tight-fitting water reflection. USEC also performed a second set of calculations with the deposit adjacent to an AQ-NCS 5.5-gallon drum containing optimally moderated UO_2F_2 solution. In this second set of calculations, the deposit was modeled as a right-circular cylinder to maximize the interaction between the deposit and the drum. In both cases, the results from the search sequence CSAS4 were verified with the validated sequence CSAS25, and the subcritical dimensions thus obtained were then converted to a subcritical mass. The safe mass was determined to be between 45 and 59% of the minimum subcritical mass, and therefore does not take double batching into account.

DAC-832-ZA1280-006 did not specify whether the spherical deposit or interaction models were used to derive the safe mass. The staff conducted confirmatory calculations that showed that the safe mass curve was based on the cylindrical deposit-with-drum model, which yielded mass values much lower than the individual sphere model. The staff also modeled the deposit in the deposit-with-drum model as a sphere in contact with the cylindrical drum, which reduced the system k_{eff}, and replaced the gapped mixed concrete-and-water reflection conditions with tight-fitting concrete-and-water reflection and full density water reflection. All calculations produced a $k_{eff} + 2\sigma$ below the TSR limit of 0.9634. Therefore, the staff considers the new safe mass curve bounding for all credible reflection conditions and geometrical configurations.

In addition, deposits may be found in the cascade equipment in close proximity to each other. Even when each deposit is less than a safe mass individually, the combination of deposits could exceed a safe mass in a given area and may require criticality safety controls for greater than safe mass deposits when neutron interactions are taken into account. Conditions A, B, and C apply to discrete portions of the cascade that could contain multiple deposits and this mass curve does not apply.

Therefore, the staff recommends approval of the proposed safe mass curve in TSR 2.4, Appendix B, for the purposes of determining entry into Conditions A and B of TSR 2.4.4.4. However, the staff does not recommend approval of the new safe mass curve for the purposes of determining entry into Condition C of TSR 2.4.4.4. In addition, because the safe mass curve does not account for double batching, it should not be applied to co-located deposits (i.e., those not neutronically isolated).

The staff therefore recommends the following condition be adopted:

> **Notwithstanding the requirements of TSR 2.4.4.4, USEC shall use the safe mass curve in TSR 2.5 Appendix B, instead of the safe mass curve in TSR 2.4 Appendix B, for determining entry into TSR 2.4.4.4 Condition C. The combined mass of all deposits in the affected equipment shall be used in making this determination.**

3.0 Nuclear Criticality Safety Review

A review was performed for specific nuclear criticality safety evaluations and approvals (NCSE/As) and for USEC's overall criticality program. The purpose of this review was to verify the adequacy of the Nuclear Criticality Safety (NCS) program to operate the plant safely at 5.5wt% ^{235}U assay. Review of the NCSE/As was necessary because the NCSEs are the evaluation that forms the safety basis for the facility and demonstrates the technical basis for the controls used for NCS. Since a 100% review of the NCSE/As is not normally conducted, the staff relied on a risk-informed/performance-based assessment of the NCS program to ensure an adequate safety basis.

3.1 Nuclear Criticality Safety Evaluations/Approvals (NCSE/As)

The staff reviewed a sampling of plant NCSE/As as part of this review, conducting a licensing review of 55 NCSE/As out of 110 (this included 35 existing NCSE/As that had previously been evaluated at 5.5wt% ^{235}U assay, along with all 20 NCSE/As modified or written specifically to support HAUP). Both groups of NCSE/As were needed for higher assay operations. During the course of the review, USEC consolidated or withdrew some of the NCSE/As, reducing the total number to 53. Five of the NCSAs, covering the seal exhaust/wet air stations, shared a common NCSE. Thus, there were a total of 48 independent evaluations reviewed as part of this amendment request.

The NCSE/As were chosen for review based on the relative risk of the processes analyzed in the NCSE/As. The documents were reviewed to determine if the NCSE/A provided an adequate safety basis to support higher assay operations or contained deficiencies requiring significant revision. A risk-informed approach was taken during the review–*i.e.*, not all observed deficiencies would trigger a determination that revision was necessary (e.g., typographical errors), but only those considered to be safety-significant. Minor documentation errors would not be sufficient to require revision, but only those where the case for an adequate safety basis could not be made from the licensing submittals. The staff defined "minor deficiencies" as those that were not in compliance with the program but which did not prevent the staff from finding a reasonable assurance of safety. "Major deficiencies" were defined as those which resulted from a failure to document an adequate safety basis, and thus prevented the staff from finding a reasonable assurance of safety. Those exhibiting such “minor” problems were classified along with those found generally acceptable for the purposes of this review.

Review of NCSE/As 310-003, 1493-30, and NCSE-045

In several NCSE/As discussed in Appendix B, the density of uranium that would accumulate in various pieces of equipment under various upset conditions was credited for NCS. For uranium in oil, there were several different bounding uranium loadings used in the various evaluations (NCSE/As 310-003, 1493-30, and NCSE-045). The NRC has dealt with this issue at both Portsmouth and Paducah. USEC submitted an amendment request for removal of the Seal Exhaust overflow controls at the Portsmouth GDP, but later withdrew this request following NRC questions on the maximum uranium loading. For uranium in oil, there are three main cases considered in the evaluations reviewed in this report: (1) slugged oil that must be stored in maximum 2.1-gallon drums; (2) Seal Exhaust pumps, which are assumed to seize due to the viscosity increase resulting from a certain uranium loading in the oil; and (3) Normetex pumps, which are assumed to trip based on filter plugging resulting from an accumulation of uranium particles in the oil. In each of these three cases, a different physical mechanism and/or condition was credited for the bounding uranium density. This explains the reason that these arguments were accepted in some cases and not in others.

In the NCSE/A 1493-30, the slugged oil is that will not flow freely under gravity, must be stored in maximum 2.1-gallon drums (freely flowing oil may be stored in maximum 5.5-gallon drums). Because slugged oil is typically stored at room temperature and normal atmospheric pressure, and the properties of uranium-contaminated oil are well-characterized under these conditions, staff found this acceptable. However, slugged oil under more extreme conditions may not have the same bounding uranium density. In 1493-30, the bounding uranium density in small vacuum pumps was not demonstrated, but the upset condition considered by USEC was found to be conservative by staff. The staff performed a calculation which showed that a more realistic upset condition modeled did not have to rely on limiting the uranium loading at all.

In the Seal Exhaust pumps (NCSE-045), a bounding uranium loading of 40 wt% was assumed. Based on the technical references docketed for the HAUP review and previous determinations during the Portsmouth review, the staff considers 40 wt% adequately bounding for the upset mechanism involving the Seal Exhaust pumps.

The Normetex pumps (NCSE/A 310-003) are of much more robust construction than the Seal Exhaust pumps, and would not be critically safe at 40 wt% ^{235}U (~36 wt% uranium loading is assumed). Moreover, bounding density is based on the loading necessary to plug the filter, which is a different physical phenomenon, and staff did not consider the assumptions that fed into this evaluation to be adequately demonstrated.

In addition to the maximum uranium loading in oil, several evaluations assumed a bounding density of uranium in other media, such as alumina or machine shavings. In these cases, the controls are designed to preclude the presence of even small quantities of uranium, and many of these operations are small batch operations (GPS-11 and 710-005) where the worst-case upset would be much smaller than the critical loading.

Additional information concerning the double contingency controls for the Normetex pumps was recently submitted to NRC by letter dated February 5, 2000 (GDP-01-0031). The staff is continuing to review this information therefore the Normetex pumps and their approval are an open issue in this preliminary CER. This issue will be resolved prior to issuance of this CER in a final format. The resolution may require a certification condition.

Results of review of NCSEs

The results of the staff's review of each of the 53 NCSE/As is documented in Appendix B. The risk categorization and result of this sampling review are summarized in Table 1 for each NCSE/A reviewed. The staff concluded that, although some NCSEs had to be modified and some physical modifications had to be made to the facility, the NCSE/As as revised are adequate to maintain safety at the Paducah Gaseous Diffusion Plant. The only exception is the NCSE related to the open Normetex pump issue.

Table 1 – Results of NCSE/A Review for HAUP

Appendix Section	NCSE/A Number & Title	Risk	Result of Review
B.1	1493-07, "Accumulation of Waste Oil"	H	Found acceptable
B.2	3971-17, "Operation and Maintenance of Cascade Compressor Seals"	H	Found acceptable
B.3	360-006, "Operation and Maintenance of the C-360 Cold Trapping System"	H	Required revision prior to HAUP. USEC eliminated the use of this area and deleted the NCSE.
B.4	3973-09, "Uranium Recovery Systems"	H	Found acceptable
B.5	3973-35, "Change out of Alumina Trap Media"	H	Required revision prior to HAUP. USEC withdrew this specific NCSE and included the operations in another NCSE which was found to be acceptable.
B.6	400-02, "C-400 Cylinder Wash Operations at the PGDP"	H	Found acceptable
B.7	400-03, "C-400 Cylinder Hydrostatic Testing Operations at the PGDP"	H	Found acceptable

B.8	GEN-010, "Removal and Handling of Contaminated Equipment from the Cascade at PGDP"	H	Required revision prior to HAUP. TSR implemented for single contingency.
B.9	GEN-012, "Handling, Transport, Storage, Disassembly & Decontamination of Small Vacuum Pumps and Datum Pumps in C-400"	H	Required revision prior to HAUP. Revision found acceptable.
B.10	GEN-10-01, "Dry Air, Nitrogen Systems for Purging Off Stream/Shutdown UF_6 Equipment"	H	Required revision prior to HAUP. TSR implemented for single contingency.
B.11	3971-28, "Operation and Maintenance of the Datum Systems and Associated Pressure Instrumentation"	H	Required revision prior to HAUP. Revision found acceptable.
B.12	GPS-01, "'00' and '000' Compressor Disassembly"	H	Required revision prior to HAUP. USEC withdrew this NCSE because they will limit the operation to less than 1 wt%. No NCSE is required for this assay limit.
B.13	1493-08, "Solid Uranium Salvage"	M	Found acceptable
B.14	GEN-037, "Remediation of NCS Violations"	H	Required correction upon next revision. Would have required revision prior to HAUP, but NCSE/A not required under NCS Program. No action required.
B.15	GEN-09, "Negative Air Machine Operation and Maintenance"	H	Found acceptable
B.16	GPS-15, "Pressure Transmitter Storage, Cleaning, Repair, and Transport"	H	Low safety significant changes which require correction upon next revision.
B.17	GPS-19, "Centrifugal Pumps"	H	Required revision prior to HAUP. TSR implemented for single contingency.
B.18	GPS-25, "Disassembly and Repair of Process G-17 Valves"	H	Required revision prior to HAUP. TSR implemented for single contingency.

B.19	GEN-15, “On-site Generation, Handling, Accumulation, Storage, Transportation, and Storage of Fissile or Potentially Fissile Waste Up to a Maximum 5.5 Weight Percent Enrichment”	L	Required revision prior to HAUP. Revision found acceptable.
B.20	GPS-06, “Repair of Cascade Expansion Joints”	M	Found acceptable
B.21	1493-33, “Sample Characterization in C-710"	L	Found acceptable
B.22	GPS-030, “Chemical Treatment of Normetex Pumps”	M	Found acceptable
B.23	400-007, “C-400 Seal Disassembly and Decontamination”	L	Low safety significant changes which require correction upon next revision.
B.24	331-001, “Operation and Maintenance of the C-331 Instrument Shop Facility”	L	Low safety significant changes which require correction upon next revision.
B.25	400-004, “Portable Container Solution Transfer to the C-400 Building, No. 5 Precipitation Tank”	L	Found acceptable
B.26	310-003, “Normetex Pumps Used for UF_6 Withdrawal”	M	Required revision prior to HAUP. Certificate condition proposed to resolve.
B.27	GPS-11, “General Machining”	L	Found acceptable
B.28	CHM-001, “C-400/C-409 Floor Drains”	H	Found acceptable
B.29	GEN-001, “General Plant Limits for Activities Performed at PGDP”	H	Required revision prior to HAUP. Revision found acceptable.
B.30	1493-30, “Handling, Transport (to and from), and Storage of Small Vacuum Pumps from Uranium Containing Systems”	M	Found acceptable

B.31	CAS-004, "C-331 Seal Exhaust/Wet Air Pump Station" 310-002, "C-310 Seal Exhaust/Wet Air Pump Station" CAS-006, "C-333 Seal Exhaust/Wet Air Pump Station" CAS-007, "C-337 Seal Exhaust/Wet Air Pump Station (up to 1.8wt% ^{235}U)" 335-001, "C-335 Seal Exhaust/Wet Air Pump Station (up to 2.0wt% ^{235}U)"	M	Required revision prior to HAUP. Revision found acceptable.
B.32	GPS-16, "Normetex Pump Maintenance and Testing"	H	Found acceptable
B.33	3973-21, "Field Decontamination"	M	Found acceptable
B.34	4151-05, "Operation of the Cylinder Weigh Scales"	M	Found acceptable
B.35	360-005, "C-360 Transfer Station"	M	Found acceptable
B.36	3973-10-14, "Relocation and Storage of Two 16" NaF Traps Found in the C-400 Receiving Booth"	M	Low safety significant changes which require correction upon next revision.
B.37	400-006, "C-400 Spray Booth"	H	Required revision prior to HAUP. Revision found acceptable.
B.38	GEN-13, "Operation and Maintenance of the High Efficiency Particulate Airborne (HEPA) System in Buildings C-310 and C-360"	M	Found acceptable
B.39	409-001, "C-409 Uranium Precipitation"	H	Required revision prior to HAUP. Revision found acceptable.
B.40	CAS-014, "Cascade Holding Drums"	M	Dropped from review - not needed for NRC's programmatic assessment.
B.41	CAS-005, "Cascade Surge Drums"	M	Found acceptable
B.42	310-006, "C-310 Cylinder Burp Station"	M	Found acceptable
B.43	CAS-002, "Operation and Maintenance of the UF_6 Cascade"	M	Required revision prior to HAUP. (USEC independently recognized the need for revision prior to licensing review.) Revision found acceptable.

B.44	3974-09, “South Bank NaF Traps”	M	Found acceptable
B.45	CAS-011, “Shutdown of Cascade with & without Inventory”	M	Required revision prior to HAUP. (USEC independently recognized the need for revision prior to licensing review.) Revision found acceptable.
B.46	3974-05, “Product Withdrawal System”	M	Found acceptable
B.47	37A-001, “C-337A Feed Station Relief Drums”	M	Found acceptable
B.48	3971-07, “Operation and Maintenance of the C-310 Tops Purge Trapping System”	M	Low safety significant changes which require correction upon next revision.
B.49	710-005, “C-710 Drain and C-712 Neutralization Pit”	H	Found acceptable. Recently revised after previous version was found inadequate.

H = High Risk **M = Medium Risk** **L = Low Risk**

The breakdown of the reviewed NCSE/As in terms of risk category and adequacy are given in the table below:

Table 2 – Breakdown of NCSE/A Review Results by Risk Category

	High	Medium	Low
Found acceptable	8	13	3
Revised for HAUP review	13	4	1
Revision required after HAUP	2	2	2

This table demonstrates that out of the 48 independent NCSEs reviewed, 18 (37.5%) were found to have sufficiently significant deficiencies that revision was necessary before HAUP could be approved. Adding those with less significant deficiencies raised (6) the proportion with observed deficiencies to 50%. The staff notes that a majority of those with major deficiencies are in the high risk category, with 13 out of 23 (~56.5%) requiring revision prior to approval of this amendment request. The staff attributes this in part to the risk-informed approach in which risk was one factor in considering whether a revision would be required.

The following table provides a summary of the two classes of NCSE/As needing revision (those involving a control or operation change and those involving documentation changes) and the ultimate resolution of NRC concerns:

Table 3–NCSE/As Requiring Revision Prior to HAUP

NCSE/A	Description of Changes
NCSE/As Requiring Physical or Administrative Changes to the Plant	
360-003, "Operation and Maintenance of the C-360 Cold Trapping System" (B.3)	Needed revision because analysis done at 5.0 instead of 5.5wt%. USEC eliminated the use of this area and deleted the NCSE.
GEN-010, "Removal and Handling of Contaminated Process Equipment from the Cascade at PGDP" (B.8)	Needed revision because it contained unreliable visual estimation of mass, equipment removal was singly contingent on moderation control, and modeled assumptions were not justified. USEC established a TSR for equipment removal activities. USEC replaced visual inspections with NDA measurement. NCSE/A was also revised.
GEN-10-01, "Dry Air, Nitrogen Systems for Purging Off Stream/Shutdown UF_6 Equipment" (B.10)	Needed revision because it contained unreliable visual estimation of mass, equipment removal was singly contingent on moderation control, and modeled assumptions were not justified. USEC established a TSR for equipment removal activities. USEC replaced visual inspections with NDA measurement. NCSE/A was also revised.
GPS-01, "'00' and '000' Compressor Disassembly" (B.12)	Needed revision because Compressor Pit found with hydraulic line that invalidated evaluation assumptions. USEC withdrew and ceased operations rather than revise.
3971-28, "Operation and Maintenance of Datum Systems and Associated Pressure Instrumentation" (B.11)	Needed revisions because system features credited for double contingency were not established as NCS Structure system or components (SSCs), and assumptions were not justified. USEC revised to add several new AQ-NCS SSCs, administrative controls.
GPS-19, "Centrifugal Pumps"	Needed revision because it did not distinguish between singly and double contingency scenarios (was only doubly contingent below a certain mass) or describe the basis for double contingency. USEC revised to describe the basis for double contingency. USEC established TSRs for moderation and spacing control.
GPS-25, "Disassembly and Repair of Process G-17 Valves" (B.18)	Needed revision because it contained singly contingent scenarios. USEC established a TSR for equipment removal activities.

310-003, "Normetex Pumps Used for UF_6 Withdrawal" (B.26)	Needed revisions because system features credited for double contingency were not established as NCS SSCs, assumptions were not justified, and control reliability was not established. USEC revised to add several new AQ-NCS SSCs. New controls on uranium loading proposed by draft certificate condition.
NCSE-045 (Seal Exhaust/Wet Air Stations) (B.31)	Needed revision because the list of controls was incorrect, which could lead to wrong control flowdown. USEC revised to clarify the list of controls.
400-006, "C-400 Spray Booth" (B.37)	Needed revision because system features credited for double contingency were not established as NCS SSCs. USEC revised to add several new AQ-NCS SSCs.
409-001, "C-409 Uranium Precipitation" (B.39)	Needed revision because system features credited for double contingency were not established as NCS SSCs. USEC revised to add several new AQ-NCS SSCs.
CAS-011, "Shutdown of Cascade with & without Inventory" (B.45)	Needed revision because singly contingent scenario had not been identified, and controls not established for moderation control on shutdown cells. USEC revised and added new Fire Protection Program controls. New TSR not needed because conditions bounded by an existing TSR.
710-005, "C-710 Drain and C-712 Neutralization Pit" (B.49)	Needed revision because basis for double contingency was not established, and features relied on for double contingency were not established as NCS SSCs. USEC revised to document basis for double contingency. USEC added a new flow monitor control as an AQ-NCS SSC.
NCSE/As Requiring Revised Safety Basis Documentation	
3973-35, "Change out of Alumina Trap Media" (B.5)	Basis for assumptions about alumina loading not substantiated. USEC withdrew because this was being consolidated with other NCSE/As which were found to be acceptable.
GEN-12, "Handling, Transport, Storage, Disassembly & Decontamination of Small Vacuum Pumps and Datum Pumps in C-400" (B.9)	Needed revision because the basis for double contingency controls was not documented. USEC revised to document basis for double contingency.
GEN-15, "On-site Generation, Handling, Accumulation, Storage, Transportation, and Storage of Potentially Fissile Waste Up to a Maximum 5.5 Weight Percent Enrichment" (B.19)	Assumptions upon which double contingency arguments were based were omitted from the most recent revision. USEC revised to address these concerns.
GEN-001, "General Plant Limits for Activities Performed at PGDP" (B.29)	Needed revision because requirements for sample validity were not clear. USEC revised to clarify these requirements.
CAS-002, "Operation and Maintenance of the UF_6 Cascade" (B.43)	Double contingency arguments were not justified. USEC revised and reverted operation to singly contingent status.

3.2 Criticality Program Evaluation

Review of criticality program

As previously noted, NRC reviewed all of the NCSEs which analyzed high risk processes, and reviewed a sampling of the NCSEs which analyzed medium risk processes. Out of the 48 NCSE/As that were reviewed, 18 (37.5%) were identified requiring significant revision (*i.e.*, revision involving more than editorial changes material to the staff's determination of adequacy) before reasonable assurance of safety could be established. Because of this, the NRC staff concluded that the NCS Program contained programmatic deficiencies.

Problems were identified with the NCSE/As produced in the 1996-1997 period and which were not modified to support HAUP. Out of the 35 existing NCSE/As reviewed, 12 were identified requiring significant revision and 6 had minor deficiencies. Of the 12 found with major deficiencies, 3 were removed from the HAUP list because the operations were either eliminated or were consolidated into other NCSEs. Out of the 13 new or modified NCSEs (17 NCSAs) reviewed, 6 needed significant revision but there were none with minor deficiencies. Both categories constitute the safety basis for the facility, and should therefore be expected to provide the same level of assurance of safety. The older NCSE/As were written to support operation at 2.75 wt% assay. The staff found that these NCSE/As were adequate for that level of operation. However, based upon operation at higher assay limits, the staff believes that the existing NCSE/A should meet the same rigorous standards as those produced under the current NCS Program due to the decrease in the safety margin. And as stated in Enclosure 2 of the October 20, 2000, CAR:

> "All existing NCSE/As currently written for higher assay operations were reviewed by USEC as part of the HAUP to determine if existing NCSE/As remained acceptable for higher assay operations. Where revisions to the existing NCSE/As were found to be necessary or desirable or where new NCSE/As were required, these NCSE/As were revised or prepared, as appropriate, in accordance with the Nuclear Criticality Safety Program requirements contained in SAR Section 5.2."

The newer NCSE/As typically met a much higher standard of documentation than the legacy NCSE/As, but still exhibited programmatic deficiencies. As stated above, the legacy NCSE/As tended to have more significant deficiencies in that they exhibited a greater lack of documentation of an adequate safety basis. The deficiencies observed in the newer NCSE/As were narrower in scope and therefore tended to require less revision than those in the older NCSE/As. However, the staff noted that the proportion of NCSE/As with significant deficiencies requiring revision was not significantly different for the legacy vs. new or revised NCSE/As.

The breakdown of the reviewed NCSE/As in terms of existing vs. new/revised NCSE/As is given in the table below:

Table 4 – Breakdown of NCSE/A Review Results by Date of Origin

	Existing	New/Revised
Found Acceptable	17	7
Revision pre-HAUP	12	6
Revision post-HAUP	6	0

The proportion of NCSE/As requiring revision prior to approval of HAUP was ~34.3% for the existing NCSE/As, and ~46.2% for the new or revised NCSE/As. The staff finds that there has been a significant improvement in the quality of documentation, as evidenced by the relative amount of revision required for the two categories. However, the staff remains concerned due to the unacceptably high revision rate of the newer NCSE/As. Although the staff required USEC to correct the deficiencies and made an ultimate determination that the current existing and revised NCSE/As are acceptable and provide a safety basis for approving the HAUP review, there are still programmatic weaknesses. Therefore, NRC requested and received commitments from USEC to correct these deficiencies. NRC will be doubling its inspections frequency in this area.

The NCSE/As were rigorously reviewed to ensure that they were technically accurate, reflected the as built condition of the facility, and that all significant hazards were evaluated and appropriate controls were identified to prevent or mitigate those hazards. The main areas that were reviewed and for which NCSE/As were required to be revised are discussed below.

Review of Unlikely Events/Natural and Credible Course of Events

SAR Section 5.2.2.1, “Adherence with ANSI/ANS Standards,” states that the NCS Program has been developed to comply with ANSI/ANS-8.1-1983. SAR Section 1.6 does not identify any exceptions to this standard. ANSI/ANS-8.1-1983, Section 4.2.1, states, in part:

> “nuclear criticality safety is achieved by controlling one or more parameters of the system within subcritical limits. Control may be exercised administratively through procedures...,by physical restraints..., through the use of instrumentation..., by chemical means..., by relying on the natural or credible course of events..., or by other means. All controlled parameters and their limits shall be specified.”

ANSI/ANS-8.1-1983, Section 4.2.3, also states in part that all dimensions and nuclear properties on which reliance is placed be verified prior to beginning operations and control be exercised to maintain them. Prior to RAC 00C044, SAR Section 5.2.2.3 described controls as including passive barriers, active engineered features, and administrative controls. Following implementation of RAC 00C044, this descriptive list of control types was expanded to include reliance on the natural or credible course of events as a stand-alone means of parameter control.

Because standard industry practice recognizes the use of passive and active engineered and administrative controls for criticality safety and these concepts are well-understood and unambiguous (but the reference to the natural and credible course of events is not), the NRC requested USEC to put additional detail in the SAR, given the ambiguity associated with the use of the natural or credible course of events as means of controlling criticality safety. In the September 14, 2000, RAI, the NRC requested USEC to describe its criteria, with specific examples, for how its reliance on the natural and credible course of events is implemented to provide adequate criticality safety control. In the October 2, 2000, response to the RAI, USEC provided the requested examples. The NRC reviewed the examples and concluded that system design features and/or administrative actions relied on to ensure the unlikelihood of an event were not identified as safety-related or controlled. That is, many of the examples that USEC believed to be examples of appropriate reliance on the natural or credible course of events were actually examples of engineered and administrative controls. Because system features relied on to prevent criticality must be identified and appropriately controlled in plant NCSE/As, USEC's intent of relying on the natural and credible course of events appeared to be taking credit for parameter control without identifying explicit controls that ensure subcriticality.

NRC and USEC held several technical discussions relating to the crediting of non-NCS plant programs in establishing requirements to prevent the initiating event of a criticality accident sequence. The specific instance of this identified by the staff involved crediting the Fire Protection Program with making the occurrence of a large fire during equipment removal activities unlikely in NCSE/A GEN-10 (Section B.8). The objective of the Fire Protection Program (*i.e.*, making large fires unlikely) was credited, because it was considered unlikely that there would ever be a change to the program that would make a large fire acceptable. However, none of the specific controls established by the Fire Protection Program were identified as NCS controls.

Irrespective of the RAC, the SAR identifies the following programmatic commitments related to the identification and configuration control of criticality safety related items:

1. SAR Section 5.2.2.8 states that if an item is relied upon for criticality safety of an operation, it will be identified as AQ-NCS and NCS approval will be required before implementing the change.

- SAR Section 5.2.2.3 states that the "NCS evaluation process involves: (1) a review of the proposed operation and procedures; (2) discussions with the subject matter experts to determine the credible process upsets which need to be considered; (3) development of the controls necessary to meet the double contingency principle; and (4) identification of the assumptions and equipment needed to ensure criticality safety."

- SAR Section 5.2.2.3 commits to the double contingency principle as stated in ANSI/ANS-8.1: Process design should in general incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible.

- SAR Section 6.3.5.3 states that the "SAR, process hazard analyses, and NCSAs are developed based on the plant's known configuration. A reasonable spectrum of postulated accidents that could potentially affect the health and safety of the public and on-site workers are identified and evaluated to determine the risk involved. These evaluations are used to identify the SSCs credited for safety and to identify the boundaries of these systems."

To resolve the two issues (*i.e.*, controlling features relied on for maintaining the natural and credible course of events, and use of programs in lieu of controls), the NRC and USEC came to agreement on revising the SAR to include the following statement:

> "Where the natural or credible course of events is relied upon in whole or in part to prevent a process condition change, the factors that influence the process must be described in sufficient detail in the NCSE as criticality safety related items relied upon for safety and programmatically controlled as AQ-NCS to ensure continued availability and reliability. For items which are established, maintained and implemented by non-NCS programs, credit for availability and reliability is established as described in SAR Section 6.3.5, Physical Plant Change Control and Configuration Management, without the necessity of establishment of additional NCS controls. For situations where the NCS-credited controls do not provide adequate assurance of availability or reliability (*i.e.*, situations where non-NCS programmatic and physical plant changes could adversely affect the intended criticality safety function of the criticality safety related items), specific administrative NCS controls will be established, maintained and implemented to ensure criticality safety."

This proposed change relied on the fact that those aspects of systems, structures, and components (SSCs) relied on for NCS will be flowed down into the Configuration Management Program as AQ-NCS items. This process was described in letter GDP-00-0235, which summarized the requirements in procedure CP2-EG-CF1030, "Q/AQ-NCS/AQ Item Identification, Documentation, and Control." Under this procedure, the NCS Organization must identify the "safety-related design criteria," or those attributes of SSCs relied on for safety, which are then controlled in the CM Program. Staff reviewed this procedure and felt it was adequate to perform this function.

Based on the above SAR change, the NRC is satisfied with the proposed changes to the SAR which addresses the above concerns.

<u>Results of Criticality Program Review</u>

The staff observed an explicit change in the NCS Program that involved documentation of the safety basis, involving the use of the new term "Safety-Related Item" (or SRI) in the NCSEs. This term is not described in Section 5.2, "Nuclear Criticality Safety," of the SAR, and therefore the staff raised questions about how these SRI controls were captured by the Configuration Management Program. Based on follow-up discussions, the staff understands SRI-class items are considered "NCS SSCs" and controlled as AQ-NCS. The staff considers this a very positive development in the newer NCSE/As, and a significant improvement in documentation of the safety basis. The staff expects that the NCSE/As that were not updated to the new standard as part of HAUP will be upgraded to identify all engineered items relied on for NCS as SRI-class items, as part of its commitment to update the older documentation.

As a result of the staff's review of the NCSE/As selected for detailed review, review of other documents (RAI responses, technical references, *etc*.) described in this report, and direct observation of plant operations and interactions with plant personnel, several programmatic deficiencies in the criticality program were identified. These are described in general terms below and specific examples are listed in Appendix A. USEC has committed to resolving these types of issues for other NCSE/As on a programmatic basis.

These issues center around documenting the basis for safety, identification of items relied on for NCS, failure to identify singly contingent scenarios, maintaining and updating NCSEs, the use of administrative controls over engineered controls, and the use of neutron absorbers.

Documenting the Safety Basis

The staff found that several NCSE/As as originally submitted did not contain a well documented safety basis, as described in Table 1. Several NCSE/As were required to be revised to identify and adequately control all of the items which were being relied on for NCS, accurately represent in-plant configurations, adequately justify the assumptions contained in the analysis, and adequately distinguish between items relied on for double contingency and those discussed to demonstrate defense-in-depth. Several of these revisions required the addition of new NCS controls or operational changes.

A well-documented safety basis is essential to the continued safe operation of the facility, to ensure that: (1) all items relied on for NCS are appropriately identified and controlled by appropriate management measures; and (2) items relied on for NCS are appropriately considered in the configuration management program so they are not inadvertently changed without appropriate NCS review.

The concern here is two-fold: (1) ensure that process characteristics, design features, and assumptions relied on for NCS remain valid under the higher assay conditions, and (2) to ensure they will continue to remain valid following approval. As discussed above, many of the older NCSE/As do not meet current documentation standards and instead rely heavily on the analyst's undocumented engineering judgement.

USEC has stated that a very detailed description of the engineering judgment is not required in the documents because all changes to these documents must be reviewed by NCS staff prior to the change being approved. Because the NCS staff are very knowledgeable in these areas, they understand the assumptions and judgment which were used in the analyses and would not allow a change which reduced the effectiveness of a control. While USEC has committed to involving NCS staff in the review of all plant changes that could impact fissile material operations, this lack of a complete and fully documented safety basis makes the review process reliant on the inherent knowledge of the reviewer, not on documented analysis which increases the potential for errors. The determination that a proposed change does or does not impact any assumptions or other items relied on for NCS is based on the analyst's engineering judgement rather than a systematic review of the technical basis.

In addition, NCSEs were found which did not identify or control all items relied on for NCS, or to distinguish between items relied on for double contingency and those discussed to demonstrate defense-in-depth. Several NCSE/As were found not to contain an adequate discussion of double contingency and were required to be revised. Specific instances of this are discussed in detail throughout Appendix B.

The staff identified three cases in which the safety basis documentation did not accurately and completely represent the as-built configuration of the facility. In NCSE/A 360-006, "Operation and Maintenance of the C-360 Cold Trapping System," the analysis was done at 5.0wt% ^{235}U assay and did not bound operations. In NCSE/A GPS-01, "'00' and '000' Compressor Disassembly," a hydraulic line was found in the Compressor Disassembly Pit while the evaluation assumed there were no sources of moderator in the pit; this introduced unanalyzed hazards. Finally, in 710-005, "C-710 Drain and C-712 Neutralization Pit," operational limits and flow conditions in the Acid Neutralization Pit did not correspond to the operation as actually conducted. Two of these cases (360-006 and GPS-01) resulted in USEC withdrawing the affected NCSE/A. The existence of plant operations that do not match the operations as described in the NCSE is a programmatic weakness.

Therefore, NRC staff has determined that a higher level of detail is required in these documents on a programmatic basis and has obtained commitments from USEC to adequately identify the items relied on for safety in the NCSEs, accurately represent in-plant conditions, adequately justify the assumptions contained in the analysis, adequately distinguish between items relied on for double contingency and those discussed to demonstrate defense-in-depth, and periodically audit the NCSE/As for adequacy. These commitments satisfy the NRC's concerns for this certification amendment.

Identification of Singly Contingent Scenarios

TSR 3.11.5 requires that any singly contingent operations must have a TSR. When there is only one process-specific barrier to criticality, it requires heightened management attention that is afforded by having a TSR. The staff reviewed the NCSE/As to ensure that there were no singly contingent accident scenarios which did not have corresponding TSRs. The staff found that several NCSE/As exhibited the existence of singly contingent accident scenarios without a corresponding TSR. SAR Section 5.2.2.3 states, in part, "There are three operations which do not meet the double contingency principle. These are product cylinder operations, operations of the enrichment cascade, and removal of large cascade equipment." TSR 3.11.5 states:

> "The double contingency principle, as described in the Safety Analysis Report, shall be used as the basis for the design and operation of processes using fissionable materials. In those instances where double contingency is not met, TSRs shall be established, implemented, and maintained to prevent criticality from occurring."

During the course of the review, however, the criticality safety staff found two examples of accident scenarios that were singly contingent were found to exist, but USEC could not identify a corresponding TSR. In these cases, this involved the existence of Planned Expeditious Handling (PEH) equipment–*i.e.*, removed cascade equipment containing greater than a safe mass of fissionable material. The singly contingent scenarios were those involving moderation or interaction upsets of PEH equipment. Although there are TSRs for equipment removal activities, they do not cover all accident scenarios and therefore do not provide adequate coverage of all situations which require TSRs. The affected NCSE/As did not identify the singly contingent accident scenarios as such, and the hazard evaluations for these scenarios were intermingled with and of the same form as the double contingency demonstration for doubly contingent scenarios. There was therefore often not a clear distinction made between the basis for singly and doubly contingent scenarios in the evaluations.

Therefore NRC staff has obtained commitments from USEC to ensure that there are no singly contingent scenarios without corresponding TSRs and that the program will prevent the occurrence of this issue in the future. These commitments satisfy the NRC's concerns for this certification amendment.

<u>Administrative controls over engineered controls</u>

During this review the staff considered USEC's preferred design approach because it is an important aspect of change control process in SAR Section 5.2.2.4, "Design Philosophy and Review":

> "Design of new fissile material equipment and processes must be approved by the NCS Section before implementation and will include the use of favorable geometry or engineered controls on mass, moderation, volume, concentration, interaction, or neutron absorption, as the preferred approach over the use of administrative controls...."
>
> "The adherence to this approach is determined during the preparation and technical review of the NCS evaluation performed to support the equipment design...."
>
> "Fissile material equipment designs and modifications are reviewed to ensure that favorable geometry and engineered controls are used to advantage. Administrative limits and controls will be implemented in NCSAs to satisfy the double contingency principle for those cases where the preferred approach cannot be met."

Following the preferred design approach will alleviate the potential for degrading the safety basis as the result of gradual replacement of reliable engineered controls with less reliable administrative controls. Administrative controls are typically more economical to implement and less restrictive on operations than engineered controls, but are more susceptible to failure. There is a much higher assurance of maintained safety when plant changes made without prior NRC approval comply with the preferred design approach stated above. In its response to the RAI (GDP-00-0235), USEC stated that "HAUP modifications, and all other SSC modifications, were designed and constructed using the preferred design approach. There were no instances where the preferred design approach could not be met. Justification of the controls specified for an operation are found within the NCSE/A that governs the operation." The response then identifies several instances in which the preferred design approach was followed, but none where this approach was not followed. From this, the staff concludes that there were no plant modifications in which new administrative controls were established.

This commitment to follow the preferred design approach, along with statements that this approach had been followed for all plant modifications, is difficult to reconcile with the staff's observation of heavy reliance on the use of administrative controls. Moreover, the staff did not observe any justification in the reviewed NCSE/As for the use of administrative instead of engineered controls. The NCSE/As do not currently contain any discussion of the alternate control systems that could have been used or why administrative controls were chosen. It was not apparent to the staff that USEC has met the intent of commitments to follow the preferred design approach in SAR Section 5.2.2.4. This is important because an approach to design in which passive or active engineered controls are preferred over administrative controls ensures that the most reliable and robust controls are chosen.

Because the current SAR requires USEC to follow the Preferred Design Approach but does not require documentation of the basis for deviation from the requirement, staff requested and received a commitment from USEC that it would add wording to the SAR which requires USEC to justify and document the basis for the use of administrative over passive or active engineered controls, for all facility changes involving new administrative controls adopted after approval of the Higher Assay Upgrade Program. This commitment gives NRC the confidence that the safety basis of the facility will be maintained.

<u>Use of Fixed Neutron Absorbers</u>

SAR Section 5.2.3.1, "Application of Parameters," states, in reference to the use of neutron absorbers:

> "When neutron absorbers are used as NCS controls, the intended distributions and concentrations under both normal and credible abnormal conditions are maintained in accordance with the requirements of the applicable NCSA. These requirements are: representative sampling of the neutron absorber, sampling at a frequency justified based on the environment the neutron absorber is exposed to, samples analyzed for all material attributes taken credit for in the NCSE, and periodic inspections of fixed neutron absorbers to ensure adequate distribution as specified in the NCSE/A. Before a neutron absorber is used, for the purposes of complying with the double contingency principle, the details specifying the neutron absorber control program shall be submitted to the NRC for review and approval.
>
> "An NCS evaluation can taken credit for the neutron absorption properties of the materials (1) added specifically for the purpose of absorbing neutrons, and (2) of construction, provided an allowance has been made for manufacturing and dimensional tolerances, corrosion, chemical reactions, and uncertainties in the neutron cross-sections."

During the course of the NCSE/A reviews, the staff encountered six instances in which the materials of construction were credited in criticality calculations. The presence of these materials (such as stainless or carbon steel) often was the determining factor in whether the system k_{eff} exceeded the TSR 3.11.4 limit of 0.9634. However, no neutron absorber control program had been submitted for these materials and there was no requirement to verify the material composition of these materials, either initially upon installation, or periodically.

Based on these findings, the staff asked USEC to clarify its position on the use of construction materials as neutron absorbers in its RAI. In Letter GDP-00-0235, USEC responded that the first paragraph above only pertained to "neutron absorbers that are used as NCS controls, which means that they are specifically added to an operation and used as a poison for one of two barriers for double contingency." The response also stated: "Before the neutron absorber can be used as an NCS control for the purposes of complying with the double contingency principle, the SAR states that the details specifying the neutron absorber control program shall be submitted to the NRC for review. This paragraph only applies to absorbers that are specifically added to an operation as NCS controls." Therefore, USEC considers neutron absorbers to be "NCS controls" that are relied on "for the purposes of complying with the double contingency principle" only if they are specifically added to the operation for their neutron absorber properties.

The RAI response further states that the second paragraph above applies only to materials of construction credited as neutron absorbers. The response further states:

> "When NCS evaluations credit the neutron absorption properties of materials of construction, the SAR does not require that the composition of the material of construction be verified prior to operation and does not require a surveillance of the material composition on a frequency determined by the environment. The SAR requires that the NCS evaluation take into account dimensional tolerances due to manufacturing, corrosion, and chemical reaction concerns."

The response does, however, go on to state:

> "If the dimensional tolerances and composition of the materials of construction is important in the calculations and relied upon in the NCSE, then the material of construction and material dimensions are identified as safety related and controlled as AQ-NCS in the configuration management program."

There were instances in NCSE/As reviewed where construction materials were explicitly modeled in calculations and were needed to meet the TSR k_{eff} limit, but were not recognized or controlled as fixed neutron absorbers.

In regard to the need for surveillance of the material thickness:

> "If the equipment material thickness could change due to a corrosive environment or due to chemical reactions, then a surveillance *may* be established as an NCS control to ensure the material thickness stays within the specified values." [emphasis added]

The staff's position is when neutron absorbers are credited in the NCSE and the dimensions could change due to a corrosive environment or chemical reactions, a surveillance requirement with frequency commensurate with the corrosion/reaction rate is mandatory to ensure safety.

The staff therefore disagrees with USEC's interpretation that structural materials can be credited as neutron absorbers without verifying the composition. Although this concern is not confined to higher assay operations, the reliance on materials of construction in models in reviewed NCSE/As, when k_{eff} is approaching the TSR limit, combined with the reduced margin of safety at the higher assay, make an accurate knowledge of the properties of these materials of construction much more safety significant than at lower assay. The staff does not consider it a reasonable interpretation to conclude that the first paragraph pertains only to materials specifically added for NCS purposes, while the second paragraph pertains only to pre-existing construction materials. The NRC interprets this passage to require that all neutron absorbers credited for NCS (consisting of both categories) must be controlled, with regard to material composition and dimensions; this may include field verification, sampling, and periodic inspections (surveillances) as appropriate. Moreover, the staff considers the distinction between materials added before and after the evaluation is irrelevant. There is no difference neutronically between the properties and behavior of a neutron absorber added specifically for NCS purposes and one added for other purposes (construction materials) and later credited for NCS. Both classes are relied on for double contingency if credited in calculations used to show subcriticality under both normal and credible abnormal conditions. USEC has discretion whether to credit neutron absorbers for NCS, and must control all pertinent properties of materials–including material composition and dimensions–that are credited for NCS, to ensure adequate safety.

Although the staff believes that this is the correct interpretation of the quoted sections of SAR Section 5.2.3.1, there is sufficient disagreement over the precise meaning to require clarification. (Note: This is applicable to any other material properties credited for NCS, such as reflectors.) Therefore, the staff requested and received a commitment from USEC to institute management controls whenever neutron absorbers are credited in meeting double contingency principle or reduces system k-effective, to ensure that: (1) the material composition of absorbers/reflectors shall be verified and adequately modeled prior to operation; (2) degradation in the material composition and dimensions such as due to corrosion or chemical reactions shall be evaluated, and a periodic inspection of the material composition and/or dimensions shall be established with a frequency sufficient maintain these properties within acceptable limits; and (3) all relevant physical properties of such absorbers/reflectors shall be identified and controlled as NCS SSCs in the plant Configuration Management Program.

Where materials of construction credited for NCS are present prior to approval to operate at 5.5wt% ^{235}U assay, and their exact material composition cannot be determined, USEC committed to determining whether they are required to maintain subcriticality (k_{eff} below the TSR limits). If the system cannot be demonstrated adequately subcritical without crediting the materials of construction, USEC committed to establish additional controls as needed to ensure subcriticality in the event that the materials of construction are not credited.

The staff recognizes that due to the extensive operating history and age of the plant, there may be areas in which it is not feasible to determine the exact material composition of structural components. In these cases, there should be sufficient other controls such that subcriticality is ensured without reliance on the presence of the neutron absorber.

This commitment will ensure that neutron absorbers are used correctly and gives NRC the confidence that the safety basis of the facility will be maintained.

4.0 Security and Safeguards Review

Review of Fundamental Nuclear Material Control Plan Changes

By letter dated October 20, 2000 (GDP 00-0168), and in conjunction with the amendment request, USEC submitted changes to its Fundamental Nuclear Material Control Plan (FNMC). Because of its nature, this plan is not publically available. The NRC reviewed the changed and issued a RAI dated December 1, 2000, which requested additional information for two material control and accounting issues. The first issue consists of the updated use of 55-gallon containers for storing uranium bearing solid and liquid materials and its inclusion in the item control program. The second issue deals with a clarification of the use of assay range codes for the facility's nuclear material database sorting purposes.

By letter dated December 22, 2000 (GDP 00-0233), USEC resubmitted the revised pages 7-8 and 8-2 of its FNMC Plan, which provided all modifications and clarifications requested by the NRC RAI letter. Upon review of the resubmitted pages of the Plan and its detailed description and justification in the attached enclosures, staff finds that USEC fully addressed the two subject issues in a satisfactory manner and the revised pages are adequate. The staff therefore concludes the performance objectives and system capabilities required by the regulations are satisfied, and the revised plan is deemed acceptable.

Physical Security and Transportation Protection

Also in letter dated October 20, 2000 (GDP 00-0168), USEC submitted minor change to its Physical Security Plan for Transportation of Special Nuclear Material of Low Strategic Significance and its Physical Security Plan. These changes involved editorial changes in one page of the Transportation Security Plan and two pages of the Physical Security Plans to remove a references to 2.75 wt% and included the shipment of 2.5-ton and 14-ton cylinders. These changes did not impact or change the procedures and measures in place to meet the NRC regulatory requirements in 10 CFR Part 76 and the revised plans are therefore acceptable.

5.0 Radiation Protection Review

On December 1, 2000, the NRC staff requested additional information on USEC's prospective assessment of dose to the public from routine operations at the 5.5 wt% enrichment level. USEC's assessment of radiation dose to the maximally exposed member of the public shows an increase from 0.011 mrem per year, which was reported in USEC's 1998 annual report to the

Environmental Protection Agency, to 0.21 mrem per year. The staff believed that this nearly 20-fold increase was not consistent with other statements in USEC's application regarding the two-fold increase in the specific activity of uranium that will be associated with the enrichment upgrade. Therefore, the staff requested additional information on the dose assessment calculation method and assumptions, which USEC provided in its December 13, 2000 response.

Upon review of this information, especially the March 31, 2000 report number KY/E-189 titled, "Effect of Operations at a Five Percent Product Assay on Paducah Gaseous Diffusion Plant Compliance With Regulatory Limits on the Radiological Dose to Members of the Public," the staff concludes that USEC used appropriately conservative assumptions in its assessment.

The key assumptions which account for most of the increase in public dose from the 1998 estimate include: (1) using maximum annual emission rates over the period from 1992-1998 as a basis for future higher assay emissions; (2) estimating an overall increase in uranium mass emissions to the atmosphere attributable to putting more process cells on-line; (3) calculating a higher specific activity of uranium that will be processed at the 5.5% assay level, and; (4) estimating emissions of transuranic radionuclides by assuming 10% of all alpha radioactivity released will be neptunium-237.

The staff agrees with USEC's explanation that this conservatively-derived prospective dose assessment is expected to bound future assessments that are based on actual operating experience. Therefore, based on the review of the application and additional information provided by USEC on December 13, 2000, the staff concludes that USEC adequately considered the radiological effects of the proposed changes on public.

6.0 ENVIRONMENTAL REVIEW

Under 10 CFR part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions," specifically 10 CFR 51.22(c)(19), "issuance, amendment, modification, or renewal of a certificate of compliance of gaseous diffusion enrichment facilities pursuant to 10 CFR part 76," is categorically excluded and neither an environmental assessment nor an environmental impact statement is warranted.

However, for this certification amendment request, the NRC staff reviewed available environmental review documentation for the PGDP that was prepared in accordance with the National Environmental Policy Act (NEPA). Available NEPA documents include site-wide environmental assessments by both the Department of Energy(1) and the United States Enrichment Corporation,(2) and an NRC environmental assessment for approving USEC's compliance plan that was associated with their initial certificate application.(3) The NRC staff conducted this review to ensure that environmental effects associated with facility changes in support of higher assay operations remained appropriately bounded by previous NEPA analyses. Upon completion of this review, the NRC staff affirmed that there are no new and significant environmental impacts associated with higher assay operations at the PGDP.

Therefore, consistent with the bases for the 10 CFR 51.22(c)(19) categorical exclusion,(4) the NRC staff finds that issuance of the Certificate Evaluation Report for higher assay operation at the PGDP will not result in any significant new environmental impact.

CONCLUSIONS:

Upon completing the compliance evaluation of USEC's amendment application, including the SAR, TSRs, NCSE/As, and FNMC plan, the staff concludes that there is reasonable assurance of safe operation of the Paducah facility at the higher enrichment of 5.5wt% ^{235}U and that the plant will operate such that public health and safety will be adequately protected, and that the common defense and security will not be endangered. Furthermore the staff determined that the application fulfills the requirements of 10 CFR Part 76. The staff recommends that USEC be issued a certification amendment in accordance with the statements and representations contained in the SAR, and TSRs. The staff recommends the following condition:

> Notwithstanding the requirements of TSR 2.4.4.4, United States Enrichment Corporation shall use the safe mass curve in TSR 2.5 Appendix B, instead of the safe mass curve in TSR 2.4 Appendix B, for determining entry into TSR 2.4.4.4 Condition C. The combined mass of all deposits in the affected equipment shall be used in making this determination.

Appendix B
NCSE/A Review

For each of the NCSE/As reviewed and discussed below, the staff reviewed the NCSE/A and then conducted a site visit to resolve any unresolved questions. The purpose of the review was to determine whether USEC had established adequate controls to ensure that the process would remain subcritical under normal and all credible abnormal conditions, and described the basis for double contingency with sufficient detail to permit an independent judgement of reasonable assurance of safety. During the course of the visit, the staff typically walked-down the operation, discussed the process with plant operations and criticality safety staff, and also reviewed additional background information upon which controls and assumptions were based. Staff also performed many confirmatory calculations to verify the models used to establish NCS limits in the NCSEs. Several of these NCSE/As resulted in Requests for Additional Information (RAIs), and several were required to be revised prior to approval of the HAUP amendment. Unique features of each NCSE/A are discussed in the following sections.

B.1 NCSA 1493-07, "Accumulation of Waste Oil"

B.1.1 PROCESS DESCRIPTION

The C-710 uranium-contaminated waste oil collection facility is used to collect potentially contaminated oil generated by the various laboratories in the C-710 Building. Sampling personnel retrieve the waste oil from the labs and pour it into one of the two maximum 5.5 gallon waste oil storage containers located in the front quarter of Room 143. The remaining 3/4 of the room is used as a 90-day accumulation area for liquid uranium salvage material, which is collected in 30-gallon containers each of which is stored in a carboy.

B.1.2 EVALUATION

The principal risk of a criticality in this operation is the handling of oil which contains greater than 30.6wt% uranium. This oil will not flow freely out of the laboratory containers due to accumulation of uranium-bearing materials. Oil which does not flow freely has reacted with enough uranium-bearing material to cause an increase in viscosity to the extent that the uranium contamination approaches 23wt% (modeled at 30.6wt% for conservatism). This is controlled by requiring any oil that will not flow with gravity to be handled under NCSA GEN-12.

The evaluation considered the potential for waste streams to become mixed due to the various containers stored in Room 143, although the evaluation considered this incredible. It was shown that even in the event of mixing the waste streams, the system would still be subcritical since pourable uranium-contaminated oil is less reactive than an equal volume of optimally moderated liquid uranium salvage. It was also shown that the system was subcritical if another container was poured on top of non-pourable oil, because the volume of the oil containers would only allow, at maximum, one container of liquid salvage to be transferred into it and the total mass of uranium held up in the liquid salvage would be insufficient to cause a criticality when mixed with the oil.

The following parameters are controlled for this operation:

a. Mass is controlled indirectly by limiting the oil allowed to be accumulated in a waste oil storage container to that which will flow freely (i.e., not slugged). Laboratory analyses show that slugged oil will not flow by gravity when the uranium contamination exceeds 23% due to the viscosity increase. This was analyzed in the evaluation as 30.6% for conservatism.
b. Enrichment is limited to 5.5 weight percent U-235.
c. Volume is controlled by limiting the waste collection containers to 5.5 gallons or less. These have been shown to be subcritical when double batched (i.e., two 5.5 gallon containers side by side are subcritical)
d. Geometry is controlled by limiting the size of containers that can be used in the room and by physically controlling the container storage holders.
e. Interaction is physically and administratively controlled on waste containers by use of angle iron mounted on the floor and minimum 4-inch spacing except when pouring oil into a container.

The staff reviewed both the evaluation and associated supporting documentation. Although it was deemed incredible in the evaluation, consideration was given to the potential for waste streams to become mixed due to the various containers stored in Room 143. It was shown that even in the event of mixing the waste streams, the system would still be subcritical since pourable uranium contaminated oil is less reactive than an equal volume of optimally moderated liquid salvage, and because if mixed in the other direction, the volume of the containers that only one container of liquid salvage could be transferred to a waste oil container and the total mass of uranium held up in the liquid salvage would be insufficient to cause a criticality when mixed with the oil. Overall, the staff considers the recommended conditions of approval given in the NCSE to be appropriate and adequate to assure that this operation meets the double contingency principle.

B.2 NCSA 3971-17, “Operation and Maintenance of Cascade Compressor Seals”

B.2.1 PROCESS DESCRIPTION

Compressor seals are used to prevent the outleakage of UF_6 and to minimize wet-air inleakage into the cascade. During normal operation of the cell, a compressor may break down or the seals will begin to degrade or fail, which necessitates changing them out. The process of changing out seals consists of taking a cell off-line and purging it of any residual UF_6, removing the seals from a compressor, transferring the removed seals in a seal can to a temporary storage area, and installation of new seals on the compressor. Next, the old seals in their seal cans are transferred from the temporary storage area to the C-400 Building for disassembly and decontamination. Once there, they are disassembled and decontaminated by hand to the extent possible. Disassembled seals are then moved to the radio-frequency furnace where they are heated to facilitate complete disassembly. Any non-reusable parts are discarded in maximum 5.5 gallon waste drums, and the reusable parts are further processed by ether dry-honing or washing in soda-ash solution to remove any remaining uranium-bearing materials on the seal parts.

The NCSE demonstrated that this operation meets the double contingency principle when the recommended conditions of approval given in NCSAs 3971-17, 3971-17-01, and 3973-27 are met.

B.2.2 EVALUATION

The principal risk of a criticality in this entire operation is the buildup of uranium in the decontamination baths or dry-honers. The dry-honers are not addressed in this evaluation (ref. NCSE 3973-33). The decon baths are maintained in a safe configuration by limiting the maximum depth of solution accumulation to a safe slab of 5.0 inches or less for an enrichment of 5.5 wt%.

The following parameters are controlled for this operation:

a. Mass is not controlled directly, but has been analyzed for the maximum loading on a seal and found to be adequately subcritical when contained in a seal can.
b. Enrichment is limited to 5.5 weight percent U-235.
c. Volume is controlled by limiting the waste collection containers to 5.5 gallons or less.
d. Geometry is controlled in the seal decontamination soda-ash baths to a slab of 5 inches or less.
e. Interaction is administratively controlled on waste containers and operators are administratively restricted to movement of one container at a time.

The staff reviewed both the evaluation and associated supporting documentation. During review of the calculational results, it was identified that one of the cases, specifically case ***5kst1900***, was not performed at the maximum 5.5 wt% enrichment, but rather 5.0wt%. Since this one particular case is bounded by the rest of the analysis, NRC did not consider it necessary to revise the NCSE prior to approving HAUP; USEC committed to address this issue in the next revision. It was also noted that there was not clear basis for the time required to hydrate a deposit, but it is known that the time required is longer than the time it takes to close up a cell to the atmosphere (ref. Letter to USEC dated Sept. 14, 2000, Paducah Higher Assay Upgrade Request for Additional Information). Overall, the staff considers the recommended conditions of approval given in the NCSE to be appropriate and adequate to assure that this operation meets the double contingency principle.

B.3 NCSA 360-006, “Operation and Maintenance of the C-360 Cold Trapping System”

B.3.1 PROCESS DESCRIPTION

Originally, the Cold Trap system was used to maintain the pressure at the sample cabinets low enough to withdraw samples and evacuate the sample container. The traps consist of two cylindrical manifolds with a volume of approximately 0.1 ft^3 and were cooled by circulating a refrigerant through coil immersed in an R-113 bath. Any trapped UF_6 would then be evacuated to the C-360 building surge drums. Suction for the traps was provided by a pair of vacuum pumps with a nominal volume of 3 quarts. To prevent contamination of the oil in the pumps, gas flow was directed through 8-inch diameter x 40-inch high alumina traps to allow the UF_6 to react with the alumina first.

Originally, the Condensate Drain Tanks were used to prevent UF_6 from getting into the sewer drain. Extremely small amounts of UF_6 leakage in an autoclave (i.e., less than 2-lbs/minute) would be detected by monitoring the steam condensate with a conductivity cell. Upon detection of high conductivity in the common steam condensate drain from each pair of autoclaves, the drain line would be automatically switched from the sewer drain to the condensate drain. The two condensate drain tanks are approximately 10-inches in diameter and 20-feet tall and hold approximately 75 gallons each.

2. EVALUATION

The Cold Trapping system is no longer operational and has been abandoned in place. The Condensate Drain Tanks are scheduled to be abandoned in place. In the NRC's first RAI to USEC (ref. Letter to USEC dated Sept. 14, 2000), the NRC relayed the commitment by USEC to cut and cap the lines leading to both systems prior to commencing operation under HAUP.

This operation was only evaluated at 5.0 wt% and not the 5.5 wt% limit allowed under HAUP, therefore, the operation has been abandoned and all lines leading to it will be cut and capped per the commitment by USEC referenced above. Since these systems were already abandoned, and since there were no future plans to use either system, USEC made the decision to cut and cap the systems prior to HAUP rather than updating the analysis for the new plant enrichment limit. Since there will be no credible path after this modification for uranium-bearing materials to get into either system, there is no further need for evaluation.

Provided that this system is cut and capped prior to commencing operation under HAUP, there is no safety concern.

NOTE: As of December 2000, a revised NCSE/A had been submitted to the NRC documenting that this system has been cut and capped. Staff does not consider further action necessary. The NRC therefore concludes that a reasonable assurance of criticality safety associated with the uranium recovery process exists at the enrichments specified in NCSE 059

B.4 NCSA 3973-09, "Uranium Recovery Systems"

B.4.1 PROCESS DESCRIPTION

Uranium bearing solutions generated from C-400 Spray Booth and C-400 Cylinder Wash operations are transferred to 20" diameter storage tanks which have been shown to be critically safe for enrichments up to 1.5 wt% ^{235}U. From the storage tanks, the solution is transferred to the Number 4 and 5 dissolver units which are not critically safe for enrichments greater than 1wt% ^{235}U.

The volume of the Number 5 dissolver unit is approximately 4500 gallons or 17000 liters. In the event that the assay in one storage tank set is greater than or equal to 1wt%, the solution must be isotopically diluted by mixing with other solutions containing less than 1wt% ^{235}U until the resulting mixture is less than 1wt% ^{235}U. The mixing process is done in storage tank sets. As an example, if the contents of tank set 1 is to be dilute with the contents of tank set 2, one half of the solution in tank set 1 is transferred to tank set 3. One half of the calculated volume necessary for dilution of tank set 1 is transferred from tank set 2 to tank set 3, and the rest of the calculated volume is transferred to tank set 1. The solution storage tanks are then recycled and sampled after dilution to verify proper assay.

The uranium storage tanks are interconnected by a piping network which allows one or more sets of tanks to be sent to either dissolver. The cumulative storage volume of the 10 storage tanks and 2 receiving tanks (6 distinct tank sets) is 4584 gallons. The solution storage tanks are constructed of 20 inch schedule 10 stainless steel. The outside diameter for schedule 10 pipe is 20.43 inches with a wall thickness of 0.213 inches. A mill tolerance of 12.5% is used to reduce the wall thickness further to bound expected manufacturing variations.

B.4.2 EVALUATION

The principal risk of criticality in this operation in this operation is the accumulation of enriched moderated uranium (i.e., uranium enriched to greater than 1wt% ^{235}U) in either the Number 4 or Number 5 dissolvers. A risk of criticality also exists in the accumulation of enriched uranium in excess of 1.5wt% ^{235}U in the Uranium Recovery storage tanks. NCSAs 400-02, 400-03 and 400-06 cover washing operations that could introduce enriched uranium into the Uranium Recovery system.

The following parameters are controlled for this operation:

1. Enrichment is limited to 1.5wt% ^{235}U in the solution storage tanks and to less than 1.0wt% ^{235}U in the Number 4 and Number 5 dissolvers.

2. Geometry is controlled by the design of the solution storage tanks which are favorable geometry for 1.5wt% ^{235}U.

Based on the staff's review of the NCSE/A, the staff concludes that the recommended conditions of approval given in the NCSA are appropriate and adequate to assure that the operation meets the double contingency principle. The NCSE is sufficiently documented to demonstrate that all credible upset conditions have been evaluated and controlled to ensure subcriticality. The NCSA adequately documents the conditions of approval based on the results of the NCSE.

B.5 NCSA 3973-35, "Change out of Alumina Trap Media"

B.5.1 PROCESS DESCRIPTION

Originally, this NCSA documented the conditions of approval for the change out of alumina trap media in the 18-inch and 24-inch diameter alumina traps located in Buildings C-331, C-333, and C-337. Contaminated trap media was transferred from the process traps to a waste drum. The transfer was achieved by placing a vacuum motor on top of a waste drum and allowing the waste drum to serve as the vacuum receptacle. The trap media consists of alumina which provides an ideal surface area for a chemical reaction between the UF_6 and water molecules present in the trap. The UF_6 molecules are present in the evacuated gases and are controlled to a maximum concentration of 50 ppm. The water molecules are present as moisture on the alumina. A grab sample was removed from the top of each waste container as they are generated. The alumina was handled as potentially fissile waste, until characterized through lab analysis and NDA techniques. The waste drums were stored temporarily in Temporary Staging Areas near the trap station.

B.5.2 EVALUATION

During the NRC review it was determined that the NCSE applicable to this operation would be superseded by NCSAs CAS-004, CAS-006 and CAS-007. In the NRC's first RAI to USEC dated September 14, 2000, the NRC provided the following comments with the expectation that similar issues associated with NCSAs CAS-004, CAS-006 and CAS-007 would be addressed. These comments included:

1. NCSA/NCSE 3973-35 considers a worst case loading to be no more than 0.35 g U/ g Alumina based on historical records. However, USEC has not demonstrated any basis for why this loading cannot be credibly exceeded. Calculations performed in Appendix B to the NCSE indicate that the most reactive alumina trap model involves a uranium loading of approximately 1.49 g U/ g Alumina. NRC, therefore, expects the revised NCSE to describe how the nature of the alumina trapping process provides adequate control of the mass parameter. If such a demonstration is not described, the NRC expects USEC to assume the maximum credible value of the mass parameter as a normal condition.

2. According to NCSE Section 1, the original intent of the evaluation was to validate design modifications for operation at 5.5wt%. In evaluating potential criticality sequences, NCSE 3973-35 scenario 1i analyzes the effect of operating an alumina trap with no alumina material present. According to the NCSE, the "analysis performed justifies a control for NCS purposes that an independent verification of the presence of alumina material and the presence of the inserted pipe for traps in C-331 and C-337 for high assay operations, in an alumina trap, be performed. The contingency is considered bounded because:

    1. Independent verification of addition of alumina to the trap is required, reducing the available volume for UO2F2/H2O to a level that can be shown to be acceptably safe.

    2. Text deleted."

    The NRC expects there to be a clear flow down from the contingency analysis to the identification of controls necessary to ensure double contingency. SAR Section 5.2.2.3 states that the NCSE process involves, in part, discussions with subject matter experts to determine the credible process upsets which need to be considered. Since the uranium loading of an alumina trap is dependant on the amount of alumina present, and the failure to install alumina is not criticality concern (criticality will not result in an alumina trap which contains no fissile material, and double contingency for the seal exhaust pump downstream of the alumina trap is not dependant on the efficiency of the alumina trap), the NRC questions why this scenario is being considered.

Since the criticality safety for change out of alumina traps would be provided under NCSAs CAS-004, CAS-006 and CAS-007, this NCSA 3973-35 was withdrawn by USEC from further HAUP reviews.

B.6 NCSA 400-02, "C-400 Cylinder Wash Operations at the PGDP"

B.6.1 PROCESS DESCRIPTION

Cylinders are placed on the cylinder wash stand using the building overhead crane. The cylinder plug is removed and the lower spray nozzle is inserted. After clean water is injected for approximately 2 minutes (8 gallons per minute), the lower spray nozzle is removed and a drain hose is attached, with the valve closed to maintain vacuum. The cylinder is rotated to the vertical position, and the cylinder valve is removed to insert the upper spray nozzle. Clean water is sprayed for approximately 3 minutes. The drain hose valve is then quickly cycled to pull any unreacted material from the plug opening into the cylinder. After sitting for approximately 10 minutes, the upper nozzle valve is opened slowly while checking for HF smoke. If smoke appears, the valve is closed, and the cylinder is rocked to the horizontal position. After approximately 5 minutes, the cylinder is returned to the vertical position and the process is repeated until no smoke appears. The cylinder is then steam cleaned using a steam wand inserted in place of the upper spray nozzle.

The drain valve is opened to drain the cylinder into the drain pan. The cylinder is purged with air, and the interior is visually inspected for deposits. If deposits are found, the cylinder is returned to the previous-described washing process.

Two 16-inch outer diameter tanks are used to store solution from the drain pan. The wash solution tanks are connected to the Uranium Recovery System transfer line while transferring solution using a flexible hose. This hose is disconnected during normal operations. The drain pan located underneath the cylinder wash stand is 8.5" deep. The superstructure above the pan has a depth of approximately 18". A depth reduction device is required to be in place while washing a cylinder.

B.6.2 EVALUATION

The principal risk of criticality in this operation in this operation is the accumulation of enriched uranium (i.e., uranium enriched to greater than 1.5wt% ^{235}U) solution in the UF_6 cylinders or storage tanks that are unfavorable for enrichments greater than 1.5wt% ^{235}U.

The following parameters are controlled for this operation:

1. Enrichment is limited to 1.5wt% ^{235}U in the solution storage tanks.

1.02 Geometry is controlled by the design of the drain pan and solution storage tanks which are favorable geometry for 1.5wt% ^{235}U.

Based on the staff's review of the NCSE/A, the staff concludes that the recommended conditions of approval given in the NCSA are appropriate and adequate to assure that the operation meets the double contingency principle. The NCSE is sufficiently documented to demonstrate that all credible upset conditions have been evaluated and controlled to ensure subcriticality. The NCSA adequately documents the conditions of approval based on the results of the NCSE.

B.7 NCSA 400-03, "C-400 Cylinder Hydrostatic Testing Operations at the PGDP"

B.7.1 PROCESS DESCRIPTION

After cleaning, UF_6 cylinders are visually inspected to ensure no visible uranium is present. The cylinders are then transferred to the hydrostatic test area and placed into cylinder saddles, where they are visually inspected prior to hydrostatic testing. Cylinders that fail visual inspection are placed back onto the cylinder wash stand for rewashing. Clean cylinders are placed into a cylinder saddle to preclude inadvertent movement during hydrostatic testing operations. The appropriate connections are made and the cylinder is filled with water. An air-operated metering pump is used to increase water pressure to the amount required. Once the hydrostatic test is complete, the water is transferred back to the storage tanks or is transferred to another cylinder.

The water storage tanks with a combined volume of approximately 1200 gallons provide the necessary containment of water utilized for cylinder hydrostatic testing operations. A transfer line is connected via flexible hose connections to the C-400 Uranium Recovery System.

The drying system for cylinders consists of radiant heaters and a dry air supply. The cylinders are heated to a temperature of approximately 150^{o} F while dry air is purged through the cylinder to aid in drying. A dew point meter is used to determine when the proper dew point (<35^{o} F) is obtained.

B.7.2 EVALUATION

The principal risk of criticality in this operation in this operation is the accumulation of enriched moderated uranium (i.e., uranium enriched to greater than 2.0wt% ^{235}U) solution in the hydrostatic storage tanks.

The following parameter is controlled for this operation:

> Mass control is provided by the requirements to bring only washed cylinders into the C-400 Hydro Area, to perform two independent visual inspections that cylinder interiors are free of

visible contamination, and to limit the number of cylinders to be hydrostatically tested to no more than 150 cylinders.

Based on the staff's review of the NCSE/A, the staff concludes that the recommended conditions of approval given in the NCSA are appropriate and adequate to assure that the operation meets the double contingency principle. The NCSE is sufficiently documented to demonstrate that all credible upset conditions have been evaluated and controlled to ensure subcriticality. The NCSA adequately documents the conditions of approval based on the results of the NCSE.

B.8 NCSA GEN-010, "Removal and Handling of Contaminated Equipment from the Cascade at PGDP"

B.8.1 PROCESS DESCRIPTION

Prior to removal from the cascade, the uranium deposit mass is quantified by use of non-destructive assay (NDA) techniques. After removal, the uranium deposit mass is verified by either a second NDA or visual inspection. Depending on the mass and configuration of the deposit, the equipment is categorized for either planned expeditious handling (PEH) or uncomplicated handling (UH). PEH categorized equipment is that which contains more than a safe mass under optimal conditions (moderation and reflection). Special handling requirements are applicable to PEH categorized equipment to limit moderator intrusion.

B.8.2 EVALUATION

The principal risk of a criticality is associated with the accident scenarios that could introduce water into equipment containing more than the minimum critical mass at 5.5wt% ^{235}U.

The following parameter is controlled for this operation:

Moderation is controlled by requirements to have equipment covered with fireproof covers and gasket seals when not in the process of being decontaminated or visually inspected [LCO 2.5.4.3]

For operations for which double contingency cannot be established, SAR Section 5.2.2.3 states that these operations are evaluated to be safe in the SAR accident analysis and that there are TSR controls in place. SAR Section 5.2.2.3 identifies the removal of large cascade equipment as a singly contingent operation. The accident analysis associated with the removal of large cascade equipment is contained in SAR Section 4.4.1. Although the SAR acknowledges sources of moderation such as fire sprinkler water and lube oil, the SAR evaluates the impact of only wet air inleakage on the basis that wet air is most consistently available. On that basis, SAR section 4.4.1 does not demonstrate that the controls established, maintained and implemented as TSRs 2.4.4.4, 2.5.4.1, 2.5.4.3 and 2.5.4.4 provide adequate assurance that the risk of criticality is low for all credible sources of moderation.

SAR Section 5.2.2.3 states that the NCSE process includes development of controls necessary to meet the double contingency principle, and the identification of the assumptions and equipment needed to ensure criticality safety. SAR Section 5.2.2.3 also states that the basis for the engineering judgement used to ascertain the independence of events and their likelihood or credibility is documented in the NCSEs. The NCSEs for singly contingent operations, therefore, document both the credibility of specific accident scenarios and the development of the TSR controls used to ensure criticality safety.

NCSE GEN-010 scenario 12 identifies an explosion of a cutting torch compressed gas cylinder. According to the NCSE, the explosion of the gas cylinder (acetylene) has a very high probability of activating the fire sprinkler system. Since the large cascade equipment being removed has the potential to be uncovered at the same time the gas cylinder explodes, the NCSE acknowledges that a criticality is possible. In its October 2, 2000, response to the September 14, 2000, NRC RAI, USEC acknowledged an acetylene cylinder explosion as a singly contingent scenario. The NRC concludes that an explosion of an acetylene cylinder and subsequent activation of fire sprinklers is a credible scenario.

NCSE GEN-010 scenario 12 states that "there are no other practical controls to preclude the occurrence of an explosion. Criticality safety of this operation must rely on worker training and knowledge in the use of cutting torch equipment..." In its October 2, 2000, response, USEC stated that TSRs 1.6.4 and 3.12 provide "sufficient defense in depth" to minimize the potential for a criticality associated with the singly contingent scenario involving an explosion of an acetylene bottle. According to USEC, TSR 3.12 requires a program to address fire hazards and the use of permits and procedures. Although the hot work procedure contains provisions for the safe placement of gas cylinders (e.g., limiting the number of acetylene cylinders in an area, securing cylinders to prevent them from being knocked over, isolating cylinders from the actual hot work), these provisions provided by USEC were not identified by the NCSE process as criticality safety related items. The availability and reliability of these procedural level provisions in preventing a criticality due to an acetylene cylinder explosion were, therefore, not assured. Since the specific provisions relied upon for criticality prevention were not specifically declared as controls, the NRC concluded that TSR 3.12 and NCSE GEN-10 did not adequately ensure criticality safety.

NCSE GEN-010 scenario 13 identifies a singly contingent scenario involving the evacuation of the area associated with the equipment removal operation. On the basis that the probability of inadvertent sprinkler activation has been calculated to be approximately 10^{-6}/yr for scenario 13, USEC concluded in its October 2, 2000, response, that "an emergency without a fire and sprinkler activation would be dealt with expeditiously and the plant would comply with the existing TSRs as soon as possible." Section 4.2 of ANSI/ANS-8.1-1983 required that criticality safety parameters be controlled within subcritical limits. According to USEC, the basis for the 10^{-6} probabilistic calculation is a quantitative fault tree involving frequency of inadvertent sprinkler activation and the frequency of sprinkler activation in any 2 hour time period. Since these frequencies are dependent on the material condition of the sprinkler system and the identification and control of activities taking place in the area which could inadvertently set off the sprinkler system (*i.e.*, heavy equipment movements, equipment cutting, *etc*.) adequate assurance that the frequencies assumed in the calculation will be maintained cannot be established without declaring these aspects as criticality safety related items. The NRC concluded that USEC's basis for the low likelihood of inadvertent sprinkler activation did not adequately justify the lack of controls to take mitigating actions such as securing the fire sprinklers to the area with the open PEH equipment.

By letters dated December 8, 2000, and January 5, 2001, USEC submitted amendment requests to modify TSRs 2.5.4.3 (addressed in NCSE GEN-010 scenario 13) and 3.12 (addressed in NCSE GEN-010 scenario 12) to address the above moderation concerns. The modifications to TSR 2.5.4.3 include steps to isolate the local sprinkler system in the event of inadvertent actuation, and to cover equipment openings with prestaged waterproof covers. The modifications to TSR 3.12 specifically identify elements that must be applied to the removal of equipment containing PEH deposits, the Product Withdrawal Area or the C-400 Decontamination Facility. The combination of these elements ensure the prevention of a fire large enough to initiate sprinkler activation which could become a source of moderation for a PEH deposit. These elements include:

1. Issue and use of welding/burning hotwork permits before the start of welding, burning or other hotwork;

2. Posting of a dedicated, continuous fire watch, equipped with portable CO_2 or dry chemical fire suppression equipment for welding, burning or hotwork operations;

3. Isolation and draining of the C-400 sprinkler system protecting the area where uncovered PEH equipment is located; and

4. Pre-fire plans addressing NCS requirements for building entry during situation evaluation and fire fighting activities.

In a separate NRC review, concerns were expressed about the exclusion of the concrete floor between pieces of PEH equipment due to the fact that the additional reflection could increase both units' reactivity, thereby underestimating the safe spacing limit of 10 feet edge-to-edge. USEC performed calculations simulating a study done in a paper titled, "Effects of Concrete Composition in Nuclear Criticality Safety Calculations," by G.R. Handley, R.C. Robinson, and J.C. Cline (ref. Transactions of the ANS, June 1990, v.61, p. 182-184.) to demonstrate that the increase in overall reactivity at the lower enrichments used at PGDP would not be adversely affected.

Also, in the original submittal by USEC, there were three methods described for quantifying the mass present in a potentially PEH piece of equipment, one of which allowed using two visual inspections to quantify the mass present. After lengthy discussions with USEC staff, it was determined that this methodology was not conservative, and USEC amended the NCSE/A to only allow visual inspections to determine if any uranium was present ("go/no go" determination), and to quantify any visually identified mass using established NDA methods (*i.e.*, at least one NDA measurement required). This is a conservative approach and is adequate to ensure proper identification of PEH equipment.

Based on the modifications made to TSRs 2.5.4.3 and 3.12, and the methodology for identifying PEH equipment via visual/NDA inspection, the staff concludes that adequate controls are in place to ensure criticality safety. The staff concludes, therefore, that PEH equipment operations may be performed safely, and complies with programmatic requirements.

B.9 NCSA GEN-012, "Handling, Transport, Storage, Disassembly & Decontamination of Small Vacuum Pumps and Datum Pumps in C-400"

B.9.1 PROCESS DESCRIPTION

Small vacuum pumps are used to supply vacuum to a large assortment of operations and equipment. Some of these uses are related to processes which have the potential of contaminating the pump oil with uranium. The oil is removed from these pumps in Building C-400. The pumps are stored on racks that contain three shelves until the oil is drained and placed in 5.5 gallon drums. Slugged oil which has to be scraped out of the pumps is placed in drums containing a maximum of 2.1 gallons. Once the oil is removed, the pumps are dismantled and the parts are decontaminated. After the pumps are clean, they may be monitored and moved to Building C-720 or are disposed as waste.

B.9.2 EVALUATION

The principal risk of a criticality in this operation is the accumulation of oil-moderated enriched uranium (i.e., uranium enriched to greater than 1wt% ^{235}U) in unfavorable geometry.

The following parameter is controlled for this operation:

1. Geometry is passively controlled by the use of favorable geometry containers (5.5 gallon and 2.1 gallon drums for free-flowing and slugged oil, respectively). Geometry is also passive controlled by specifically restricting the types of pumps that may be handled per this NCSA.

2. Geometry is administratively controlled by requiring fissile control areas (FCAs) be established where geometry restrictions are necessary for criticality safety. Inside FCAs, only containers having volumes less than or equal to 5.5 gallons may be employed.

In the September 14, 2000, request for additional information, the staff identified two technical concerns that required NCSE revisions. These concerns were:

1. NCSE scenario 9 did not clearly identify the second control relied upon for double contingency. Scenario 9 involved the draining of the pump oil into a thirty or a fifty-five gallon drum. According to the NCSE, double contingency is established by the bringing of a container greater than 5.5 gallons into the area which is required to be a fissile control area, and the placing of waste in the large container rather than the 5.5 gallon drum.

2. NCSE scenario 10 did not clearly identify the second control relied upon for double contingency. Scenario 10 involves the draining of slugged oil into a 5.5 gallon drum instead of a 2.1 gallon drum required by the NCSE. According to the NCSE, criticality is possible if the slugged oil is allowed to settle. Thus, the NCSE required controls to ensure that the 5.5 gallon drum containing slugged oil be promptly identified before the settling is allowed to come to completion into a critical

configuration. However, the NCSE did not identify the prompt identification as a control. Thus, scenario 10 as described is potentially singly contingent. Based on technical discussions held with USEC staff during the week of September 5, 2000, the scenario 10 accident sequence is no longer considered credible.

By letter dated September 29, 2000, USEC committed to revising NCSE scenarios 9 and 10 by October 30, 2000. Based on the staff's review of the revised NCSE/A, the staff concludes that the recommended conditions of approval given in the NCSA are appropriate and adequate to assure that the operation meets the double contingency principle. The NCSE is sufficiently documented to demonstrate that all credible upset conditions have been evaluated and controlled to ensure subcriticality. The NCSA adequately documents the conditions of approval based on the results of the NCSE.

B.10 NCSA GEN-10-01, "Dry Air, Nitrogen Systems for Purging Off Stream/Shutdown UF_6 Equipment"

B.10.1 PROCESS DESCRIPTION

Prior to removal of cascade equipment, the uranium deposit mass is quantified by use of non-destructive assay (NDA) techniques. After removal, the uranium deposit mass is verified by either a second NDA or visual inspection. Depending on the mass and configuration of the deposit, the equipment is categorized for either planned expeditious handling (PEH) or uncomplicated handling (UH). PEH categorized equipment is that which contains more than a safe mass under optimal conditions (moderation and reflection). Special handling requirements are applicable to PEH categorized equipment to limit moderator intrusion.

When the cell or line containing the deposit is evacuated of UF_6 it is purged with dry air to just below atmospheric pressure to allow the operators to cut into the system. This ensures that there will not be sufficient moisture in the cell to cause a criticality.

B.20.2 EVALUATION

The principal risk of a criticality is associated with the accident scenarios that could introduce water into equipment containing more than the minimum critical mass at 5.5wt% ^{235}U.

The following parameter is controlled for this operation:

> Moderation is controlled by requirements to ensure that less than 1300 ppm of moisture is present in enriched cells being cut.

Based on the staff's review of the NCSE/A, the staff concludes that the recommended conditions of approval given in the NCSA are appropriate and adequate to assure that the operation meets the double contingency principle. During the NRC review of this NCSA, two technical issues arose involving the exclusion of the concrete floor between pieces of PEH equipment, and USEC's use of visual inspections for uranium mass determination. Although these issues were identified during the review of NCSA GEN-10-01, these issues were applicable to NCSE GEN-010 which contains the technical basis for both NCSA GEN-010 and NCSA GEN-10-01. The concrete and visual inspection issues are discussed in the section on NCSA GEN-010 (Section B.8). For the portion of NCSE GEN-010 pertaining to the use of the dry air and nitrogen systems, the NCSE is sufficiently documented to demonstrate that all credible upset conditions have been evaluated and controlled to ensure subcriticality. NCSA GEN-10-01 adequately documents the conditions of approval based on the results of the NCSE.

B.11 NCSA 3971-28, "Operation and Maintenance of the Datum Systems and Associated Pressure Instrumentation"

B.11.1 PROCESS DESCRIPTION

Instrumentation at various points in the cascade monitor the process system and automatically regulate the pressures. The pressures in the cascade are sensed by a differential pressure transmitter and compared to a reference pressure provided by the datum systems. The datum systems use static lines to maintain controlled reference pressures of dry air to the stage differential blind multiplier transmitters. The datum systems are equipped with dry air feeds, exhaust systems, controllers and pressure transmitters for maintaining the datum pressures. Three datum systems are used: unit datum (high and low), cell datum and the freezer/sublimer datum. Since a failure of a unit high datum system in a cascade facility could result in an immediate shift of inventory, the potential exists for cell integrity to be compromised and allow moderation to enter the cascade. To minimize this potential, controls have been established to require at least two independent and unlikely failures: the loss of air supply and the loss of alarms and lock-in devices.

B.11.2 EVALUATION

The principal risk of a criticality in this operation is associated with intrusion of moderator into the cascade as a result of breaches in cell integrity due to large shifts in cascade inventories.

The following parameter is controlled for this operation:

> Moderation is controlled as it relates to prevention of a unit high datum failure and subsequent introduction of moderator into the cascade. Controls have been established to ensure that affected portions of the cascade are shut down in response to a loss of plant air before a cascade shift in inventory can occur.

In the October 6, 2000, request for additional information, the staff identified three technical concerns that required NCSE revisions. These concerns were:

1. NCSE did not provide an adequate basis for the maximum uranium loading in a single datum pump;

2. NCSE did not identify the datum pumps, pressure blind switch number one or valve E as criticality safety related items; and

3. NCSE did not demonstrate that the loss of plant air would not result in a shift in cascade inventory.

By letter dated October 27, 2000, USEC adequately responded to the first two concerns by committing to revise the NCSE specifically to address the identification of criticality safety related equipment relied upon for double contingency. USEC's response to the third concern, however, was inadequate, and NRC notified USEC by letter dated November 21, 2000, that it provide an adequate response by December 4, 2000. By letter dated December 4, 2000, USEC adequately responded to the third concern by committing to shut down cells affected by a decrease in plant air pressure below that required for pneumatic instrumentation control.

Based on the staff's review of the revised NCSE/A, the staff concludes that the recommended conditions of approval given in the NCSA are appropriate and adequate to assure that the operation meets the double contingency principle. The NCSE is sufficiently documented to demonstrate that all credible upset conditions have been evaluated and controlled to ensure subcriticality. The NCSA adequately documents the conditions of approval based on the results of the NCSE.

B.12 NCSA GPS-01, "'00' and '000' Compressor Disassembly"

NCSE/A GPS-01 had been previously identified by USEC as having discrepancies between the safety basis documentation and the as-built system configuration. Due to the identification of previously unanalyzed sources of moderator (*i.e.*, a hydraulic line in the Compressor Disassembly Pit), this operation had been limited procedurally to disassembly of equipment with less than 1.0wt% ^{235}U assay in the Pit. Following discussions with the Paducah NCS staff during the NRC licensing site visit, USEC made the determination to remove NCSE/A GPS-01 from higher assay operations until the NCSE/A could be revised to reflect the actual conditions present in the compressor Pit.

B.13 NCSA 1493-08, "Solid Uranium Salvage"

B.13.1 PROCESS DESCRIPTION

The solid salvage vent hood in Building C-710, Room 21 is used to collect solid uranium salvage for disposal. The majority of this material is the unused portion of trap mix samples and precipitate from lab analyses. Trap mix samples come from the chemical traps located in various cascade process buildings. The samples of precipitate come from the uranium recovery systems in C-400 and C-409 or from the trap mix.

There is a hood is used to handle the solid uranium salvage waste described above. A grinder which is used to homogenize solid salvage samples prior to characterization is also located in the hood. The ventilation system evacuates any vapors generated in the hood.

B.13.2 EVALUATION

The chemical trap mix and sample remainders are collected in 5.5 gallon drums located in the hood. These have been shown to be subcritical when double batched (i.e., two 5.5 gallon containers side by side are subcritical). The hood also has a sample jar table if a sample needs to be taken for the material which is being disposed. These sample jars are limited to a maximum height of 4.5 inches and a maximum inner diameter of 4.25 inches and cannot be stacked on top of one another. The number of sample jars is limited by the size of the table. This arrangement of sample jars was also shown to be subcritical in an analysis which assumed an infinite slab 4.5 inches high.

The following parameters are controlled for this operation:

a. Enrichment is limited to 5.5 weight percent U-235.
b. Volume is controlled by limiting the liquid and waste containers to 5.5 gallons each.
c. Geometry is controlled by limiting the sample jar and chemical trap sizes and disallowing stacking of containers. It is also indirectly controlled due to the fixed geometry of the grinder.
c. Interaction is administratively controlled by requiring a 2-foot edge-to-edge spacing between waste containers and all other equipment which may contain fissile material, except as specified in the NCSE. Interaction is also indirectly controlled by the fixed spacing of the grinder and sample table relative to the waste drums.

The NRC notes that an assumption used in scenario 9 concerning the buildup of material in the vertical portion of the duct was not substantiated. However, the model assumed that the 12-inch (30-cm) diameter duct contained a 7-cm annular coating of optimally moderated, fully reflected fissile material at the most reactive density. This is considered bounding given the material handled and the operations performed in the hood. Based on the staff's review of the NCSE/A, the staff concludes that the recommended conditions of approval given in the NCSA are appropriate and adequate to assure that the operation meets the double contingency principle. The NCSE is sufficiently documented to demonstrate that all credible upset conditions have been evaluated and controlled to ensure subcriticality. The NCSA adequately documents the conditions of approval based on the results of the NCSE.

B.14 NCSA GEN-037, "Remediation of NCSA Violations"

NCSE/A GEN-037 was developed to provide guidance for the remediation of NCSA violations that resulted in the loss of double contingency or the loss of a control used for a singly contingent operation. The NRC staff originally had a concern with the presence of so-called "TBD" (To Be Determined) parameter limits listed in the tables of identified accident scenarios and concluded that in order to address the stated purpose of addressing situations that fell outside the envelope of any other specific evaluation, these values should be added to the tables. Upon further consideration, it became apparent that this document did not meet the programmatic requirements of an NCSA and should not be considered an NCSA at all.

According to SAR Section 5.2.2.3, there are only three operations which do not meet the double contingency principle. These consist of product cylinder operations, operation of the enrichment cascade, and removal of large cascade equipment. Contrary to the SAR, this NCSA addresses potential operations outside the bounds of double contingency. In addition, TSR 3.11.5 requires that for instances where double contingency is not met, a TSR shall be established, implemented, and maintained to prevent criticality from occurring. Contrary to this TSR, there is no established TSR for remediation of NCSA violations.

SAR Section 5.2.2.3 goes on to describe that in emergency situations, CNS expertise or guidance is needed immediately to avert the potential for a criticality accident. In lieu of these SAR and TSR requirements, it would appear that this NCSA either needs to be specifically added to both the SAR and TSR, or this NCSA should be removed and its contents preserved in the form of a reference document available to Paducah NCS staff for the remediation of NCSA violations. While NCSE/A GEN-037 does contain information valuable to the Paducah NCS staff in responding to emergency situations, it does not meet the requirements of an NCSA under the NCS Program. Upon further discussions with USEC, the staff believes that there is no regulatory requirement to have an NCSA to cover remediation of accident or emergency conditions. Therefore, the staff concluded that it was not necessary to revise GEN-037 to support HAUP. The NCSE/A process should not be used to produce documents other than those meeting all the requirements of an NCSE/A.

B.15 NCSA GEN-09, “Negative Air Machine Operation and Maintenance”

B.15.1 PROCESS DESCRIPTION

Nuclear Power Outfitters Negative Air Machines (NAMs) are used at Paducah to remove both airborne particulates (radioactive and nonradioactive) and hazardous fumes from the work area. This minimizes personnel exposure and hazardous work areas. The NAMs are portable and can therefore be used throughout the plant.

There are two sizes of NAMs used at Paducah, the 1,000 and 2,000 cubic feet per minute (cfm) units. Both units use the same size filters, but the 2,000 cfm unit uses two parallel filter arrays identical to the 1,000 cfm filter array to allow twice the air flow. The NAM filtration array consists of three filter banks including pre-filters, media filters, and High Efficiency Particulate Air (HEPA) filters. The NAMs are also equipped with a fire detection system that will shut off the blower and isolate the unit if the air stream reaches a preset temperature.

B.15.2 EVALUATION

Since the NAMs may be used throughout the plant under a variety of different conditions, the fissile material loading is unknown. Therefore, USEC assumed the maximum possible loading on the filters. This was determined by performing loading studies which ascertained the maximum thickness of material that could be captured on a filter before the filter plugged. The staff reviewed the filter loading study and considers the mass determination adequately conservative. The amount of material that can accumulate on the filters was shown to be less than half of the subcritical limit of 49.1 lbs U given the operating controls on the vacuum pressure and usage, and filter types allowed.

The following parameters are controlled for this operation;

a. Mass is controlled by requiring removal of any loose material during each filter change and by requiring that NAMs cannot be left unattended under fissile solution lines. Mass is indirectly controlled because the amount of material that can accumulate on the filters is less than half of the subcritical limit of 49.1 lbs U.

b. Enrichment is limited to 5.5 wt% U-235.

c. Volume is controlled by requiring inspection of the NAM interior each time the filters are changed.

d. Interaction is administratively controlled by requiring a 2-foot edge-to-edge spacing between batches of filters and all other equipment/containers which may contain fissile material. The NAMs are also required to maintain this spacing during storage. Two NAMs, each with the maximum fissile material loading on the filters stored side-by-side, were shown to be subcritical.

e. Concentration/density of the fissile material is limited due to the filter construction and the NAM blower and system capacity.

USEC has limited the operation of the NAMs such that the maximum possible fissile material loading is less than the subcritical mass limit of 49.1 lbs of U, fully reflected and optimally moderated. USEC also performed calculations to determine the reactivity of violating the spacing requirements for the NAMs and filters. USEC's calculations demonstrated that two NAMs stored side by side were subcritical with optimum moderation and reflection. Also, an infinite array of filters was shown to be subcritical. Thus, this operation was shown to meet the double contingency principle provided that the required controls listed in the NCSE are followed.

B.16 NCSA GPS-15, "Pressure Transmitter Storage, Cleaning, Repair, and Transport"

B.16.1 PROCESS DESCRIPTION

Pressure transmitters are used at Paducah to monitor pressures throughout the plant. Some of these transmitters are used in processes which may contaminate the pressure transmitter primary with uranium. The transmitters are taken to Building C-720, where they are stored until they are taken to Building C-409 for cleaning. The transmitters are cleaned one at a time in the pressure transmitters cleaning station by forcing water through the transmitter. Although there is not much fissile material in a single transmitter, the cleaning water is used to clean several transmitter before it is disposed of in 5.5 gallon containers. After cleaning, the transmitters are tagged to distinguish them as clean, then stored in Building C-409. When needed, they are sent back to Building C-720 for calibration and testing.

B.16.2 EVALUATION

The cleaning station is limited to 10 collection cylinders; each has a maximum volume of 350 milliliters and holds the water used to clean the transmitter. Only one transmitter is cleaned at a time. The maximum volume of the transmitter is 300 milliliters. Thus, because the station is limited to 10 collection cylinders at 350 milliliters each and one transmitter at 300 milliliters, the maximum volume where fissile material may accumulate is limited to 3.8 liters. This is conservative and acceptable since it takes more than 19 liters of water to make UO_2F_2 go critical at this enrichment.

Once the solution in the collection cylinders is dirty it is placed in one of two liquid and waste containers which are limited to 5.5 gallons each. These have been shown to be subcritcal when double batched (i.e., two 5.5 gallon containers side by side are subcritical).

The following parameters are controlled for this operation:

a. Enrichment is limited to 5.5 weight percent U-235.
b. Volume is controlled by limiting the total transmitter primary volume per batch to 5.5 gallons during cleaning and storage, liquid and waste containers are limited to 5.5 gallons each.
c. Interaction is administratively controlled by requiring a 2-foot edge-to-edge spacing between batches of transmitters that have not been cleaned, waster containers and all other equipment which may contain fissile material. Liquid and waste containers must also meet this spacing requirement.

During the licensing review and on-site licensing visit, the staff identified several minor inconsistencies in the NCSE that required revision. The staff determined that these inconsistencies did not affect the ability to determine that the operation as described was adequately safe; therefore, the staff identified GPS-15 as requiring revision upon the next update to address these concerns. The inconsistencies noted were:

1. The NCSE states that a volume of greater than 5.44 gallons is considered a criticality concern, but allows the use of 5.5-gallon drums in areas with sprinkler coverage. Following questions by the staff, USEC acknowledged that the 5.44 gallons should have been referenced as a subcritical limit in the NCSE but was not; the 5.5-gallon limit actually was imposed from the analysis in GEN-15 (Section 19).

2. The NCSE contains an inconsistency in the number of collection cylinders required to constitute an unsafe volume–one section states that 80 collection cylinders are needed, and another section states that only 58 collection cylinders are needed. USEC acknowledged that the 80 cylinder reference should have been removed from an earlier version.

3. The NCSE states that the minimum critical mass of UO_2F_2 is ~22.3 kg, but based on the dimensions given, the cleaning station is capable of holding slightly more than this mass of uranium and water upon sprinkler activation. Therefore, it was not apparent why the system was considered subcritical under upset conditions involving sprinkler activation. USEC acknowledged that this mass limit of 22.3 kg was only for the uranium (not the uranium-water mixture) and was a subcritical mass limit, not the minimum critical mass as stated.

Based on the staff’s review of the NCSE/A and subsequent discussions with the NCS engineers, the staff concludes that the recommended conditions of approval given in the NCSA are appropriate and adequate to assure that the operation meets the double contingency principle. The NCSE demonstrates that all credible upset conditions have been evaluated and controlled to ensure subcriticality. The NCSA adequately documents the conditions of approval based on the results of the NCSE.

B.17 NCSA GPS-19, "Centrifugal Pumps"

B.17.1 PROCESS DESCRIPTION

GPS-19 covers the disassembly and storage of contaminated centrifugal pumps used in the UF_6 enrichment cascade. The removal of centrifugal pumps (compressors) from the cascade is covered under the NCSE/A GEN-10 (Section B.8). The pumps are classified as UH or PEH based on NDA measurements before entry into the C-400 Building. Decontamination of pumps following removal of any large deposits is covered by NCSE/As 400-006 (Section B.37) and 3973-21 (Section B.33). Thus, this NCSE/A only covers receipt and storage of pumps containing UH or PEH deposits, pump disassembly, and removal of large deposits. The material removed is stored in maximum 5.5-gallon drums in secondary containment pans.

B.17.2 EVALUATION

The main criticality hazard for this operation involves an upset with pumps containing PEH (greater than safe mass) deposits, through the introduction of water moderation or spacing upsets. The primary upset of concern is the introduction of moderator–although USEC had not demonstrated adequate subcriticality for two PEH-bearing pumps with a spacing upset, staff believes that criticality is nonetheless very unlikely without the introduction of liquid water moderation.

The following parameters are controlled for this operation:

1. Mass is not directly controlled for NCS purposes, but the equipment is characterized as UH or PEH based on NDA measurements while still covered by GEN-10. Different handling requirements are applied based on the UH or PEH status of the pumps.
2. Enrichment is limited to the plant-wide limit of 5.5wt% ^{235}U assay.
3. Volume of waste containers is limited to 5.5-gallon drums.
4. Moderation is controlled by requiring fireproof covers on openings, a fireproof tarp during disassembly operations, and by time limits for equipment decontamination.
5. Interaction is limited by requiring a minimum 2-foot spacing between waste containers (fixed by the containment pans), and administrative limits to maintain a 2-foot spacing between pumps containing more than 5 lb uranium and other equipment, a 6-foot spacing between PEH-bearing pumps and other equipment, and 10-foot spacing between two pieces of PEH-bearing equipment.

During this review, the staff raised concerns with the double contingency discussion for certain accident scenarios. In particular, the NCSE did not contain a detailed description of the two barriers needed to meet the double contingency principle for the following accident scenarios: 2, 3, 4, 6, 10, 11, and 13. The double contingency discussion for each scenario identified above is summarized below:

> Scenario 2 involved a fire in C-400 that could introduce moderation to the equipment through sprinkler activation. The staff had a concern with this scenario in that the time limit for decontamination was not stated, and there was no technical basis provided.

Scenario 3 involved a spacing upset between two pumps, but did not provide a technical basis for the 2-foot spacing credited or demonstrate the subcriticality in the event of a spacing violation.

Scenarios 4 and 13 involved the rearrangement of material into a criticality configuration inside the pump; although moderation was required for criticality, the unlikelihood of rearrangement of material or the adequacy of the time limit in making this unlikely was not demonstrated. Scenario 13 in addition relies on the geometry of “a typical centrifugal pump casing”, but the dimensions of the pump casing are not credited as an NCS SSC.

Scenario 6 relied on requiring that waste be disposed of in maximum 5.5-gallon drums, and states that procedures and training ensure that the controls in NCSA 3973-21 (Section B.33) will be adhered to. However, the effect of an upset involving the use of an unsafe volume container was not evaluated.

Scenario 10 involved moderation from oil, and merely asserted that there is insufficient oil to allow criticality (centrifugal pumps do not contain an oil reservoir). The discussion assumes that there was an insufficient oil film on the outside of the pumps to cause criticality, but the amount needed for criticality was not evaluated.

Scenario 11 involved the use of a nearly-full 5.5-gallon drum in decontaminating more than one pump. For UH deposits, this is still subcritical because the safe mass determination takes double batching into account (*i.e.*, safe mass is <50% minimum critical mass). In addition, UH deposits are bounded by the interaction of two 5.5-gallon drums with the worst case uranium loading. For PEH deposits, the double contingency argument is based on the requirement that only empty drums be used to initiate decontamination. However, the effect of violating this requirement was not evaluated.

The staff could not readily determine that double contingency had been met for these scenarios, either because: (1) the two barriers preventing criticality were not described in sufficient detail to permit their independent verification; or (2) the technical basis for controls and limits was not provided. In addition, the scenarios as written did not describe the effect of the upset condition on the system or demonstrate that the system would remain subcritical under credible abnormal conditions. Instead, a defense-in-depth argument justifying why the upset would not occur was presented.

Therefore, the staff identified this as requiring revision prior to approval of HAUP. In USEC’s RAI response dated October 27, 2000 (Letter GDP-00-0188) and in discussions during the on-site licensing visit, USEC acknowledged that the scenarios discussed were only truly doubly contingent for UH deposits, and these scenarios were singly contingent for PEH deposits. That is, maintaining less than a safe mass was the first barrier to criticality (for UH deposits only) and the controls on moderation, spacing, and decontamination time limits were credited as the second control. The NCSE/A did not make this distinction clear.

USEC committed to revise the NCSE/A to demonstrate adequate double contingency for UH deposits and describe the singly contingent status for PEH deposits. The NRC did not review the revised NCSE/A in detail due to time constraints, and instead defers this review to future inspection efforts. However, in its review of the original scenarios, the staff believes there is adequate safety for UH-bearing equipment.

With regard to PEH-bearing pumps, this issue was resolved by the TSRs for moderation and spacing that are discussed in detail in Section B.8 (for NCSE/A GEN-10). The feature of these TSRs unique to this process is that the sprinkler system in the C-400 Building must be valved out during the time that PEH-bearing equipment is uncovered. The RAI response also states that the technical justification for the time and interaction limits is provided in NCSE GEN-10.

Based on these TSRs for PEH-bearing pumps and the double contingent arguments described for UH-bearing pumps, the staff finds reasonable assurance of safety in the disassembly and decontamination of contaminated centrifugal pumps in the C-400 Building. **Review of the revised NCSE/A remains an open issue that can be deferred to the inspection function.**

B.18 NCSA GPS-25, “Disassembly and Repair of Process G-17 Valves”

B.18.1 PROCESS DESCRIPTION

Process G-17 valves are used at Paducah in the cascade. These valves may also be brought in from Portsmouth and K-25. However, valves from off-site are not covered by this NCSE. These valves are either Crane or Bayard valves from 4 to 42 inches in size. The valves are disassembled and decontaminated in Building C-400, and then taken to Building C-720 where they are repaired and reassembled. This NCSE only covers the operations performed in Building C-400 and C-720. The removal of G-17 valves from the cascade is covered in NCSE GEN-10.

B.18.2 EVALUATION

Due to the large size of some of these valves, there is a potential to acquire a large mass of fissile material. Therefore, as described in NCSE GEN-10, the valves are determined to either require uncomplicated handling (UH) or Planned Expeditious Handling (PEH) depending on the mass of fissile material. This is determined prior to transport to Building C-400. The valves are handled in Building C-400 according to their classification as UH or PEH. The PEH valves require very stringent handling requirements due to their potential of having an unsafe mass of fissile material. PEH valves must have all openings covered to keep out water and strict time limits require that the fissile material mass is reduced to a safe mass in a timely manner.

The valves are disassembled and decontaminated in Building C-400. The valve bellows is visually checked for wear and damage, then pressure checked to evaluate its integrity.

Damaged bellows are removed. Material removed from the valves is treated as fissile material and placed in maximum size 5.5 gallon containers. An NDA is then performed on the valve bellows to ensure that any uranium in the bellows is less than 5 pounds. The valves/valve components are then moved into Building C-720 for repair and assembly. Reassembly operations may include material buildup welding, grinding, and machining. Any waste generated in Building C-720 from this operation is assumed to contain fissile material and is disposed of in maximum size 5.5 gallon containers.

The following parameters are controlled for this operation:

a. Enrichment is limited to 5.5 weight percent U-235.
b. Volume is controlled for the waste generated by limiting the waste containers to 5.5 gallons each.
c. Geometry of the G-17 valves is fixed by design. Valves that are 4 inches or less are considered safe by design.
d. Moderation is controlled by requiring PEH valves to have all openings covered, use of a fireproof tarp over the valve during disassembly, and time limits for decontamination.
e. Interaction is administratively controlled by requiring a 2 foot edge-to-edge spacing between waste containers and all other equipment which may contain fissile material. Process G-17 valves designated as UH with greater than 5 lbs of uranium are also required to meet the 2 foot edge-to-edge spacing. Only one PEH valve is allowed in the area at a time and must maintain a 6 foot edge-to-edge spacing from other fissile material.

Two scenarios, Scenario 2 (loss of moderation control) and 3 (loss of interaction control), do not meet the double contingency principle for PEH valves. By letters dated December 8, 2000, and January 5, 2001, USEC submitted amendment requests to modify TSRs 2.5.4.3 (NCSE GEN-010 scenario 13) and 3.12 (NCSE GEN-010 scenario 12) to address the above moderation concerns. By letter dated November 17, 2000, USEC submitted amendment requests to add a new TSR 2.5.4.5 to address the interaction concerns. Based on the modifications made to TSRs 2.5.4.3 and 3.12, the methodology for identifying PEH equipment via visual inspection, and the addition of TSR 2.5.4.5, the staff concludes that adequate controls are in place to ensure criticality safety. The staff concludes, therefore, that PEH equipment operations may be performed safely, and complies with programmatic requirements.

B.19 NCSA GEN-15, "On-site Generation, Handling, Accumulation, Staging, Transportation, and Storage of Fissile or Potentially Fissile Waste Up to a Maximum 5.5 Weight Percent Enrichment"

B.19.1 PROCESS DESCRIPTION

The purpose of this NCSE/A was to provide a technical basis for the nuclear criticality safety (NCS) of handling waste generated from higher assay operations (as described in GEN-15) and the operation of Temporary Fissile Storage Areas (as described in WM-01). Wastes are collected from operational activities conducted in a Fissile Control Area (FCA) and removed from process equipment. The wastes are placed in waste drums approved by NCS. The waste is transported and staged in a Potentially Fissile Accumulation Area (PFAA) until the drums are filled. Once the potentially fissile drums are full, they may be taken to a Temporary Staging Area (TSA) for transport to a Temporary Fissile Storage Area (TFSA) for characterization or they may be taken from the PFAA straight to a TFSA. If the waste was not sampled upon removal from the process, analytical samples are drawn from the containers while they are in the TFSA. Based on the results of a drum monitoring analysis and one analytical sample analysis, or two analytical samples when the drum monitor is unavailable, homogenous waste is characterized to determine the need for NCS controls. If the waste cannot be sampled (i.e., heterogeneous waste), the characterization is based on a single drum monitor analysis. Depending on the NCS spacing requirements for the waste, it is transported to a long-term storage facility using NCS-approved handling practices. If the sample analysis and/or drum monitor results indicate that NCS spacing controls are not required, the waste from that container may be consolidated with wastes of similar type from other containers into another NCS-approved container. All waste to be consolidated must satisfy a consolidation criterion by NCS before it can be combined with other waste in an accumulation container. The consolidation criterion is a fissile mass limit above which a container cannot be consolidated in an accumulation drum. The accumulation container has a maximum safe mass limit on its contents below which NCS spacing controls will not be required. This accumulation container is analyzed for uranium content using the drum monitor to confirm that the uranium content is less than the consolidation criterion.

This NCSE demonstrated that this operation meets the double contingency principle when the conditions of approval in NCSAs GEN-15 and WM-01 (now combined as NCSA KY/S-253 (Rev. 5)) are met.

B.19.2 EVALUATION

The combinations of successes and failures of packaging, sampling, labeling, and transporting waste was modeled by using four event tree models (i.e., two for handling fissile waste, one for handling fissile waste as NCS-exempt waste, and one for handling waste that does not require NCS controls). The possible end states were: (1) no violation, (2) violation due to fissile waste stored as NCS-exempt waste, (3) violation of NCS spacing requirement (i.e., either one drum full reflection or three drums together), (4) violation due to critical configuration during transport, and (5) violation due to mass of consolidation drum exceeding safe mass.

USEC identified three scenarios not covered by the event tree models: (1) use of the 5-inch high waste drum containment pans, (2) storage area drains and tunnels, and (3) sample placement on top of a waste drum. USEC identified these three as meeting double contingency.

In addition, USEC performed criticality calculations using the 5.5-gallon drum for credible abnormal conditions for storage of the drums (i.e., single waste drum, two drums side-by-side (with various water reflection conditions), two drums side-by-side in the corner of two concrete walls, two drums stacked (with various water reflection conditions), infinite array with varying interstitial moderation, 40 x 40 array (with varying uranium concentration and full outside reflector), 40 x 40 array with spacing upset conditions).

The following parameters are controlled for this operation:

1. Mass is not controlled directly, but is used to determine the need for spacing requirements. Drums containing less than 120 grams ^{235}U may be exempted from NCS spacing requirements. Mass is utilized as a criterion for the long-term storage of waste.

2. Enrichment is limited to 5.5 wt.% ^{235}U.

3. The generation and storage of potentially fissile waste is limited to maximum 5.5-gallon capacity drums. Non-pourable potentially fissile waste oil is limited to maximum 2.1-gallon capacity drums.

4. Geometry is controlled for the 2.1, 5.5, 35, and 58-gallon waste drums; the physical restraint system; and containment pans.

5. Moderation is not controlled. Slugged oil (i.e., oil that does not flow freely) is controlled by storing it in 2.1-gallon drums based on optimum conditions of moderation for the uranium and oil mixture.

6. Interaction is administratively controlled. Drums can be handled and transported one at a time unless using the physical restraint system. There is a 2-foot edge-to-edge spacing requirement for drums with at least 120 grams U-235.

Based on the accident analysis in NCSE Section 5 and NCS contingency analysis in Section 6, USEC determined that the generation, handling, accumulation, and staging of potentially fissile waste drums will remain subcritical under both normal and credible abnormal conditions. The USEC determined accident analysis (*i.e*., over 150 sequences) showed that several unlikely failures must occur to result in a critical configuration. Furthermore, the USEC NCS calculations showed that single spacing violation of one drum at full reflection, and spacing violations of three drums, and handling violations of a fissile drum in the low level waste area are subcritical. USEC determined that no single credible events were identified which could lead to an inadvertent criticality; and therefore double contingency was satisfied. USEC determined that the double contingency principle would be met for this operation given that the required controls given in the NCSE were followed.

The staff reviewed the NSCA/Es and associated supporting documentation. The staff reviewed the assumptions behind and the input probabilities to the event tree methodology, as well as the training of the USEC staff to use the methodology. The staff agrees with the three scenarios that are not criticality concerns. The staff reviewed the storage k-effective calculations and determined that they are acceptable. The staff did identify two concerns that resulted in GEN-15 having to be revised prior to approval of HAUP, however. These two concerns were:

a. During a modification to GEN-15 using an Engineering Change Form (ECF), several NCS assumptions were dropped from the evaluation.

b. There was a mismatch between the page numbers included in ECF 01-KY/S-253 and the NCSE. This meant that the replacement pages in the ECF did not completely correspond to the original evaluation.

The NCSE was subsequently revised to address these concerns. Overall, the staff considers the recommended conditions of approval given in the NCSEs to be appropriate and adequate to assure that this operation meets the double contingency principle.

B.20 NCSA GPS-06, "Repair of Cascade Expansion Joints"

B.20.1 PROCESS DESCRIPTION

Prior to removal from the cascade, the uranium deposit mass is quantified by use of non-destructive assay (NDA) techniques. After removal, the uranium deposit mass is verified by either a second NDA or visual inspection. Depending on the mass and configuration of the deposit, the expansion joint is categorized for either planned expeditious handling (PEH) or uncomplicated handling (UH). PEH categorized equipment is that which contains more than a safe mass under optimal conditions (moderation and reflection). Special handling requirements are applicable to PEH categorized equipment to limit moderator intrusion.

B.20.2 EVALUATION

The principal risk of a criticality is associated with the accident scenarios that could introduce water into equipment containing more than the minimum critical mass at 5.5wt% ^{235}U.

The following parameter is controlled for this operation:

> Moderation is controlled by requirements to have equipment covered with fireproof covers and gasket seals when not in the process of being decontaminated or visually inspected [LCO 2.5.4.3]

Based on the modifications made to TSRs 2.5.4.3 and 3.12, and the methodology for identifying PEH equipment via visual inspection (reference CER for NCSA GEN-010), the staff concludes that adequate controls are in place to ensure criticality safety. The staff concludes, therefore, that PEH equipment operations may be performed safely, and complies with programmatic requirements.

B.21 NCSA 1493-33, "Sample Characterization in C-710"

B.21.1 PROCESS DESCRIPTION

The purpose of this NCSE/A was to provide technical justification for sample characterization of fissile/potentially fissile materials, and provide the operational limits and conditions necessary to ensure an adequate margin of safety. Fissile/potentially fissile material is sampled to identify the actual uranium assay and/or content of the material. The fissile material samples come from a variety of processes in buildings C-331, C-333, C-335, C-337, C-310, C-360, C-710 laboratory facility, C-400 and C-409.

NCSE/A 1493-33 contains no operational controls other than those controls related to establishing the acceptability of the sampling process. NCSE/A GEN-01 contains the controls relied upon for demonstrating double contingency associated with the taking and handling of samples.

B.21.2 EVALUATION

The principle risk of criticality is associated with a failure to establish the double contingency principle due to lack of independence between sample measurements used to establish double contingency. SAR Section 5.2.2.3 states that the double contingency principle may be applied with at least two controls on a single parameter provided “the violation or failure scenarios of the controls shall be independent.” A failure to ensure independence of controls could result in a complete loss of controls due to a single failure (common mode failure).

The sample determination process requires that two samples be taken from a source and analyzed in an independent manner. However, due to limited equipment and personnel, the same instrument and laboratory personnel may be used. Where such analytical results are used to demonstrate the unlikeliness of two concurrent events required for double contingency, independence of the two events may not be assured without the implementation of specific controls. NCSE Section 4.4 develops the specific controls associated with the use of single instruments for analyzing NCS dual samples. These controls include periodic quality assurance checks on the instrumentation and requirements for performing multiple samples on separate days or by separate technicians.

The parameters controlled for criticality safety of sampling operations are contained in NCSE/A GEN-01 and NCSE/A GEN-08. No specific parameters are identified per NCSE/A 1493-33.

Based on the staff’s review of the NCSE/A, the staff concludes that the recommended conditions of approval given in the NCSA are appropriate and adequate to assure that the operation meets the double contingency principle. The NCSE is sufficiently documented to demonstrate that all credible upset conditions have been evaluated and controlled to ensure subcriticality. The NCSA adequately documents the conditions of approval based on the results of the NCSE.

B.22 NCSA GPS-030, “Chemical Treatment of Normetex Pumps”

B.22.1 PROCESS DESCRIPTION

NCSE/A 310-003 provides the technical justification for operation of the Normetex pumps, including identification of safety related items. The boundary for NCSE/A GPS-30 begins with the GPS-30 requirement to rotate the pump hand crank at least two revolutions. If the pump cannot be rotated by hand, NCSA GPS-30 prohibits chemical treatment or recovery operations on the pump.

Chemical recovery is required when chemical contaminants such as arsenic provide excessive loading or binding in the scroll region of the pump. Such loading or binding can cause pump failures resulting in a potentially unsafe situation. To facilitate removal of the chemical contaminants, moist air is drawn through the pumps to liquify and vaporize these compounds. After an acceptable leak rate and a dry air purge/evacuation, treatment gas mixtures are released into the pump. After varying lengths of time, the gas mixtures are evacuated to a sample/test buggy.

B.22.2 EVALUATION

The principal risk of a criticality in this operation is the accumulation of oil-moderated enriched uranium (i.e., enriched up to 5.5wt% ^{235}U) in unfavorable geometry (volumes).

The following parameter is controlled for this operation:

1. Waste drums/containers must meet the requirements of NCSE/A GEN-15 (internal volume less than or equal to 5.5 gallons).

2. Sample buggy component volumes must meet the requirements of NCSE/A GEN-6

3. Individual trap volumes must meet the component and batch requirements of NCSE/A 3971-12 (total volume of batch must be less than 10 liters).

Based on the staff's review of the NCSE/A, the staff concludes that the recommended conditions of approval given in the NCSA are appropriate and adequate to assure that the operation meets the double contingency principle. The NCSE is sufficiently documented to demonstrate that all credible upset conditions have been evaluated and controlled to ensure subcriticality. The NCSA adequately documents the conditions of approval based on the results of the NCSE.

B.23 NSCA 400-007, "C-400 Seal Disassembly and Decontamination"

B.23.1 PROCESS DESCRIPTION

Compressor seals are used to prevent the outleakage of UF_6 and to minimize wet-air inleakage into the cascade. During normal operation of the cell, a compressor may break down or the seals will begin to degrade or fail, which necessitates changing them out. The process of changing out seals consists of taking a cell off-line and purging it of any residual UF6, removing the seals from a compressor, transferring the removed seals in a seal can to a temporary storage area, and installation of new seals on the compressor. Next, the old seals in their seal cans are transferred from the temporary storage area to the C-400 Building for disassembly and decontamination. Once there, they are disassembled and decontaminated by hand to the extent possible. Disassembled seals are then moved to the radio-frequency furnace where they are heated to facilitate complete disassembly. Any non-reusable parts are discarded in maximum 5.5 gallon waste drums, and the reusable parts are further processed by ether dry-honing or washing in soda-ash solution to remove any remaining uranium-bearing materials on the seal parts.

Since this operation involves fissile material, an NCSE was performed by USEC to demonstrate that this operation meets the double contingency principle when the recommended conditions of approval given in NCSA 400-007 are met.

B.23.2 EVALUATION

The principal risk of a criticality in this entire operation is the buildup of uranium in the decontamination baths or dry-honers. The dry-honers are not addressed in this evaluation (ref. NCSE 3973-33). The decon baths are maintained in a safe configuration by limiting the maximum depth of solution accumulation to a safe slab of 5.0 inches or less for an enrichment of 5.5 wt%.

The following parameters are controlled for this operation:

a. Mass is not controlled directly, but has been analyzed for the maximum loading on a seal and found to be adequately subcritical when contained in a seal can.
b. Enrichment is limited to 5.5 weight percent U-235.
c. Volume is controlled by limiting the waste collection containers to 5.5 gallons or less.
d. Geometry is controlled in the seal decontamination soda-ash baths to a slab of 5.0 inches or less.
e. Interaction is administratively controlled on waste containers and operators are administratively restricted to movement of one container at a time.

For every potential criticality scenario, a double contingency analysis, contained in Section 4.1 of the NCSE, was performed by the licensee. USEC determined that all of these scenarios met the double contingency principle given that the required controls given in the NCSE are followed. Staff reviewed both the evaluation and associated supporting documentation.

Case SC-10, which looks at the disruption of seals while in storage, referenced “random factors” as preventing a criticality. Upon further review of this case, adequate controls supporting the double contingency principle for this scenario are in place. Discussions with USEC indicated that this random factor reference was intended to show defense in depth. USEC agrees that this is misleading and has committed to modifying the NCSE to remove the reference to the "random factors" in the evaluation of Case SC-10 to better reflect their double contingency basis.

During review of the calculational results, it was identified that one of the cases, specifically case ***5kst1900***, was not performed at the maximum 5.5 wt% enrichment, but rather 5.0wt%. Since this one particular case is bounded by the rest of the analysis (because it was part of a parameter study that yielded a maximum k_{eff} ~ 0.6), the NRC did not consider it necessary to revise the NCSE prior to approving HAUP; USEC committed to address this issue in the next revision. It was also noted that there was not clear basis for the time required to hydrate a deposit, but it is known that the time required is substantially longer than the time it takes to close up a cell to the atmosphere (ref. Letter to USEC dated Sept. 14, 2000).

During a reference check of GAT-225, which was used to authorize a maximum slab depth of 5.0 inches, it was noted that Table I stated that a 5.0 inch slab was only safe for 5.0 wt% assay material and not 5.5 wt% as the evaluation states is the maximum enrichment limit. After discussion with USEC staff and further review of GAT-225, it was determined that the actual referenced portion of GAT-225 was Figure A-10, which provides the estimated minimum critical water-reflected slab depths for various enrichments and indicates that with the associated conservatism built into the soda-ash bath (for which the 5-inch maximum depth is applicable), this is an adequate margin of safety. Since this reference was available in GAT-225, and is conservative, NRC does not require this issue to be resolved prior to HAUP. However, USEC has agreed to commit to address this issue during the next evaluation update and will clearly state how this reference is used and the associated conservatism built into the evaluation for the maximum slab depth allowed in the evaluation. (Ref. Letter to USEC dated Nov. 1, 2000, Paducah Higher Assay Upgrade Request for Additional Information).

Overall, the staff considers the recommended conditions of approval given in the NCSE to be appropriate and adequate to assure that this operation meets the double contingency principle.

B.24 NCSA 331-001, "Operation and Maintenance of the C-331 Instrument Shop Facility"

B.24.1 PROCESS DESCRIPTION

Operations in the C-331 Instrument Shop Facility include: (1) the sandblaster, (2) chemical trap handling, (3) chemical trap mix change, (4) vacuum pump handling, (5) valve repair, (6) instrument repair, and (7) HEPA vacuum cleaners. The sandblaster is used to clean rust and other non-fissile deposits from the equipment. The sandblaster is designed like a glovebox, with the sandblasting equipment added. The bottom of the glovebox has a metal grating, with the inlet to a cyclone separator vacuum system below the grating. The cyclone separator separates the sand from the high velocity air stream and recycles the sand.

Chemical traps enter the C-331 Instrument Maintenance Shop as part of instrument buggies brought into the shop to collect gas drawn as a sample before being drawn into the vacuum pumps. Vacuum pumps and traps are routinely handled as part of the operation of the instrument buggies.

B.24.2 EVALUATION

The principal risk of a criticality in this operation is associated with the long-term accumulation of enriched uranium (i.e., uranium enriched to greater than 1wt% ^{235}U) in unfavorable geometry.

The following parameter is controlled for this operation:

> Uranium mass is limited by requirements to prohibit introduction of equipment which has been exposed to a process upset until decontamination has been performed.

In the December 6, 2000, request for additional information, the staff identified two technical concerns that required NCSE revisions. These concerns were:

a. The NCSA permits operation of the sandblaster with equipment potentially containing fissile material but does not specify either a surveillance to determine the uranium loading in the sand or a time frame in which the sand is replaced regardless of uranium loading.

b. Although the NCSA precluded the introduction of line recorders subjected to a major process upset into the C-331 Instrument Maintenance Shop, the NCSA did not establish the necessary controls to reliably comply with that requirement. Specifically, the NCSA did not require that affected line recorders be documented for identification and tracking purposes. The lack of such a control reduces assurance that line recorders that have been subjected to a major process upset would not be mistakenly introduced into the C-331 Instrument Maintenance Shop.

By letter dated December 26, 2000, USEC adequately responded to the staff’s concerns. In particular, USEC committed to revising the NCSA to preclude process-gas equipment from the sandblaster. In addition, USEC committed to crediting an existing line recorder NCSE which demonstrates double contingency based on line recorder design (i.e., assumes the tubing, traps and pump components of the line recorder are completely filled with fissile material). Since an adequate technical basis for the line recorders already exists, NRC does not require that this NCSA be modified prior to issuance of the HAUP amendment, and accepts a commitment from USEC to address these issues during the next evaluation update.

Based on the staff’s review of the revised NCSE/A, the staff concludes that the recommended conditions of approval given in the NCSA are appropriate and adequate to assure that the operation meets the double contingency principle. The NCSE is sufficiently documented to demonstrate that all credible upset conditions have been evaluated and controlled to ensure subcriticality. The NCSA adequately documents the conditions of approval based on the results of the NCSE.

B.25 NCSA 400-004, “Portable Container Solution Transfer to the C-400 Building, No. 5 Precipitation Tank”

B.25.1 PROCESS DESCRIPTION

The purpose of this NCSE/A was to provide a technical basis for the nuclear criticality safety (NCS) of transferring waste material from portable containers directly into the C-400 No. 5 Dissolver/Precipitation Tank. Waste material is transferred from portable containers directly into the C-400 No. 5 Dissolver/Precipitation Tank. The waste streams consist of both uranium and non-uranium bearing solutions. Aqueous non-fissile solutions from various plant sources are poured or pumped into the C-400 No. 5 precipitation tank. NCSA 3973-09 (Apr. 1996) demonstrated subcriticality for the 4,500 gallon No. 5 precipitation tank for all uranium bearing solutions with assay less than 1.0 wt.% U-235. NCSA 3973-09 required two independent samples of transfers from non-portable tanks (i.e., receiving tanks and storage tanks) to be less than 1.0 wt.% U-235 and required the resulting maximum enrichment in No. 5 precipitation tank to be less than 1.0 wt.% U-235. Therefore, transfers from portable containers must also demonstrate to either have less than 1.0 wt.% U-235 or demonstrate that the solutions are not associated with a uranium bearing process.

Because this operation involves fissile material, NCSE 009-00 was performed by USEC to demonstrate that this operation meets the double contingency principle when the conditions of approval in NCSA 400-004 are met.

B.25.2 EVALUATION

From a criticality safety perspective, the material of concern is a uranium-bearing solution in the that could contain material at or above 1.0 wt.% U-235. Normal operations require either two independent samples of less than 1.0 wt.% U-235 according to CP2-EG-NS1034, “NCS Requirements for Sample Labeling, Handling, Chain of Custody, and Assay Smears”, or documentation that the waste came from a waste stream identified as chemicals under Table 2 of GEN-07, which is exempt from NCS control in accordance with CP2-EG-NS-1033. In addition, because potentially fissile material waste is stored in 2.1-gallon drums, 5.5-gallon drums, and 21-liter poly bottles, USEC determined that a requirement for this operation would be that transfers to the No.5 precipitation tank would not be allowed using those types of containers or any container with a Fissile or Potentially Fissile label or marking. The NCSE/A uses administrative controls and one engineered feature (i.e., a cover plate to the access port of the tank) to prevent greater than 1.0 wt.% U-235 material from being transferred to the tank from portable containers.

The following parameters are controlled for this operation:

c. Enrichment is controlled to less than 1.0 wt.% U-235. This is the only parameter controlled to ensure that the operation remains subcritical under all credible normal and abnormal conditions because the No. 5 tank is not favorable geometry and no other parameters will be controlled.

d. Volume is not specifically controlled. However, 2.1-gallon drums, 5.5.-gallon drums, and 21-liter poly bottles are prohibited from being portable containers that can be transferred to the No. 5 precipitation tank.

e.

USEC stated that no single event initiators were identified that would result in an inadvertent criticality. Sixteen different process upsets were identified for this operation. Of these, two scenarios were deemed not to be a criticality concern, four scenarios were identified as a criticality concern that had been addressed in other NCSE/As, and twelve scenarios were identified as a potential criticality concern. For the twelve potential criticality concern scenarios that had not been addressed in a previous NCSE/A (i.e.,NCSE/A 1493-33, Rev.2; NCSE/A GEN-01, Rev. 4; NCSE/A GEN-07 in KY/S-253, Rev. 4) a double contingency analysis was performed by USEC and contained in Section 4.2 of NCSE 009-00. USEC determined that all of these scenarios met the double contingency principle given that the required controls given in the NCSE are followed.

The staff reviewed the NCSE/A and associated supporting documentation. The staff agrees with the two scenarios that are not criticality concerns. For the twelve scenarios not evaluated in other NCSEs, the staff reviewed the double contingency analysis in Section 4.2. The only thing being controlled is to keep the enrichment of transferred material less than 1.0 wt.% U-235. This is done by an engineered feature (i.e., a cover plate over the access port of the tank) and administrative controls (i.e., dual independent sampling, use of a tamper indicating device, use of identification on containers, access control to the access port of the tank, prohibition against the use of potentially fissile/fissile containers). Overall, the staff considers the recommended conditions of approval given in the NCSE to be appropriate and adequate to assure that this operation meets the double contingency principle. (The description of the arguments used by USEC to demonstrate that double contingency is met are questionable, however, the recommended conditions of approval in the NCSE are sufficient to meet double contingency.)

B.26 NCSA 310-003, “Normetex Pumps Used for UF_6 Withdrawal”

B.26.1 PROCESS DESCRIPTION

NCSE/A 310-003 covers the operation and maintenance of Normetex pumps used in the UF_6 product withdrawal system. These are large geometry positive displacement pumps that intake UF_6 at a pressure of approximately 7 psia and eject the gas to the withdrawal condensers at approximately 30 psia, maintaining the UF6 in the gas phase throughout normal operations. These pumps are designed with recirculating lube oil loop containing an oil pump and two large geometry oil reservoirs, and recirculating cooling water to cool the oil. The interior of the pump contains a moving and a fixed spiral scroll that are part of the UF_6 containment barrier and prevents mixing of UF_6 and oil in the unfavorable geometry oil reservoirs. The gas is moved in the region between the scrolls as the moving scroll rotates, and is expelled through the outlet pipe in the center of the scroll region.

B.26.2 EVALUATION

The staff is continuing its evaluation of this NCSE/A. It is an open issue in this preliminary CER .

B.27 NCSA GPS-11, "General Machining"

B.27.1 PROCESS DESCRIPTION
As process equipment in the cascade fails or is removed for repair or maintenance, it is decontaminated either in the field or in specialized equipment located in Building C-400. Following decontamination, surveys are conducted to verify equipment surfaces to be machined are below a transferrable contamination limit of 50,000 dpm/100 cm^2 beta/gamma.

When equipment arrives at C-720, another survey is performed to verify the contamination level to less than 50,000 dpm. Machining equipment involved in this operation include large milling machines, lathes, drill presses, grinders and other cutting equipment. Operations with this equipment could result in removal of material from process equipment parts containing fissile material enriched to 5.5wt% ^{235}U. The removed material includes metal chips, shavings, or grinding dust that accumulates in the catch basins beneath the machining equipment or on the machine shop floor. The fissile material removed from process equipment components as a result of machining will consist of any remaining transferrable and fixed contamination. Machining waste is removed from the machining equipment on a daily basis and placed into maximum size 5.5 gallon drums.

Machining operations on contaminated or potentially contaminated items will be performed without the use of the machine recirculating cooling system.

B.27.2 EVALUATION

The principal risk of a criticality in this operation is associated with accumulation of enriched uranium (i.e., uranium enriched to greater than 1wt% ^{235}U) in unfavorable geometry.

The following parameters are controlled for this operation:

a. Uranium mass is controlled through the requirements for decontamination followed by verification by survey prior to machining. Accumulation of mass during machining operations is controlled by the requirement for daily cleanup.

b. Machining waste is collected in maximum size 5.5 gallon drums and removed from the machining area.

Based on the staff's review of the NCSE/A, the staff concludes that the recommended conditions of approval given in the NCSA are appropriate and adequate to assure that the operation meets the double contingency principle. The NCSE is sufficiently documented to demonstrate that all credible upset conditions have been evaluated and controlled to ensure subcriticality. The NCSA adequately documents the conditions of approval based on the results of the NCSE.

B.28 NCSA CHM-001, "C-400/C-409 Floor Drains"

B.28.1 PROCESS DESCRIPTION

The purpose of this NCSE/A was to provide a technical basis for the nuclear criticality safety (NCS) of the floor drain and outside storm drain systems that are potentially affected by the operations (*i.e.*, decontamination activities) in buildings C-400 and C-409 involving significant quantities of fissile solution (*i.e.*, ~1200 gallons) with uranium solutions at enrichments up to 5.5 wt.% U-235, as well as the solution lines that connect those operations. (This NCSE/A does not provide a technical basis for the NCS of the operations in those buildings because those bases are found in the NCSE/A for that equipment.)

After solutions are generated in the buildings, they are characterized by both U-235 enrichment and uranium concentration. USEC determined that the only operations that process large volumes of potentially fissile solution are the C-400 Cylinder Wash, Stokes Pump Disassembly, Vacuum Pump Disassembly, Spray Booth, Spray Booth Tanks, Seal Disassembly, C-409 Solution Storage, C-409 NaOH Precipitation System, C-409 Filtrate Tanks, and C-400/C-409 Transfer Line.

(Note that this is really not an NCSE/A, it is a hazard identification document used to provide NCS basis information concerning a hazard (i.e., inadvertent criticality in the drains systems) that was not identified in the NCSE/As specific to the equipment in the buildings. Rather than update the individual NCSE/As, USEC decided to create this one document that evaluates a specific hazard common to all the equipment in the two buildings. In the cases where this NCSE/A declines to evaluate an accident sequence, that accident sequence must be addressed in the NCSE/A for that equipment.)

B.28.2 EVALUATION

Because the floor drain and storm drain systems are not favorable geometry, the control against an inadvertent criticality must occur at the access points to the floor drain and storm drain systems from the C-400 and C-409 buildings, which are the drains and sumps in the two buildings. To prevent an inadvertent criticality via the drain systems, USEC uses administrative controls and engineered controls (i.e., sealed all but two of the drains and filled all the sumps in the two buildings), as well as maintenance and surveillance activities. The two drains remained unsealed because they are required for the operations in the buildings. To prevent an inadvertent criticality via those two unsealed drains, USEC uses administrative controls and engineered controls (i.e., placed a 5-inch high collar on one drain and a 6-inch high collar on the other drain), as well as maintenance and surveillance activities. To prevent an inadvertent criticality during a transfer, USEC uses administrative controls, as well as maintenance and surveillance activities.

The following parameters are controlled for these operations:

1. The maximum enrichment for the C-400 cylinder wash is 2.0 wt.% U-235. The C-400 uranium recovery system is limited to 1.5 wt.% U-235 at the storage tanks and the #4 and #5 dissolvers are limited to less than 1.0 wt.% U-235. The maximum enrichment for the rest of the operations is 5.5 wt.% U-235.

2. Volume is limited based on the storage tank capacities associated with each operation. In most of the operations, the floor pans are designed to contain the maximum volume of solution in the tanks.

3. Geometry is controlled by limiting the accumulation depths of containment pans to the corresponding safe slab depths for the maximum enrichment of the operation.

4. The concentration is controlled in the C-400 uranium recovery system storage tanks. The concentration is limited to less than 100 gU/l for any solutions that have uranium enrichments greater than or equal to 1.0 wt.% U-235 (up to 1.5 wt.% U-235). Solution transfers to C-409 uranium recovery system are limited to no more than 100 gU/l unless they contain less than 5.5 g U/l (with a maximum of 300 g U/l). Dual independent sampling according to procedure CP2-EG-NS1034 is used.

5. Interaction is fixed by design for storage tank spacings and other potentially fissile material such as waste drums are spaced a minimum of two feet edge-to-edge.

USEC determined that the double contingency principle could be met for all the hazards provided that the controls, criticality safety related items, maintenance activities, and surveillance activities described in the NCSE are included in the NCSA and adhered to during decontamination operations.

The staff reviewed the NCSE/A and associated documentation and determined that (1) the hazards that were identified were appropriate and (2) the administrative controls, engineered controls, maintenance activities, and surveillance activities that were identified are sufficient to meet the double contingency principle for the hazards identified. Overall, the staff considers the recommended conditions of approval given in the NCSE to be appropriate and adequate to ensure that these operations meet the double contingency principle. (The NCSE/A is not well organized, the NCSE/A is confusing as to the actual scope of the document, and the NCSE/A requires the NCS Engineer to know that there are multiple documents that describe the overall NCS basis for the operations in the C-400 and C-409 buildings.)

B.29 GEN-001, “General Plant Limits for Activities Performed at PGDP”

B.29.1 PROCESS DESCRIPTION

The purpose of GEN-001 is to describe requirements necessary to ensure independence of dual sampling and instrumentation measurements (Non-Destructive Assay, or NDA) for double contingency. GEN-001 contains general requirements for sampling and measurement methods throughout the plant, but does not establish specific requirements due to the number and variety of plant conditions and operations. During on-site discussions with plant personnel, the staff recognized that this NCSE/A is not intended as a stand-alone document, but is used in conjunction with an NCSE/A for a specific plant operation. GEN-001 covers the in-field sampling of samples taken to determine mass, concentration, or enrichment, labeling, and the transportation of samples, but does not cover laboratory analysis. Specific requirements to ensure that the samples are representative and provided in plant operating procedures. This NCSE/A also contain several requirements for the in-plant transportation of samples received from off-site, violation of NCS limits, and containers that could collect solution leaks.

B.29.2 EVALUATION

There are two classes of criticality hazards considered–those involving criticality safety for samples that are produced in the field and transported as a batch, and those involving criticality hazards to the process relying on the independent sampling. The latter class would result from common-mode failures of the sampling program, which would compromise the independence of the samples. Samples are treated as potentially fissile (PF) until characterized as either fissile or NCS exempt; PF/fissile units must be spaced a minimum of two feet from all other PF/fissile material. Materials containing less than 1wt% ^{235}U assay or less than 15g ^{235}U are considered NCS exempt. Due to the small volume in a grab sample, the staff does not consider it credible to collect sufficient mass in a compact enough geometry to cause an accidental criticality.

The following parameters are controlled for handling and transportation of sample batches:

1. Enrichment is limited to the plant-wide limit of 5.5wt% ^{235}U assay.
2. Volume is not directly controlled, but sample batch limits are based on sample volume.
3. Interaction is controlled by administratively limiting the spacing between PF/Fissile containers.

The main criticality hazards are associated with the processes relying on dual independent sampling or measurements for NCS control. GEN-001 requires that, whenever possible, the material being sampled will be recirculated or mixed between samples, must be representative of the material being sampled, and must be verified by a second individual. Staff had several concerns that were raised during this review, which are discussed below.

The NCSE/A contains definitions for independent and representative samples, that contain less than the total requirements necessary to ensure independence and representativeness. More detailed requirements are spelled out in process-specific NCSE/As and elsewhere in GEN-001, but using these definitions could lead to less than adequate administrative measures being applied. The staff requested that this NCSE/A be revised to define these terms more accurately, and found the revision acceptable. During its on-site review, the staff compared the requirements for independent sampling in GEN-001 and NCSE/A 400-006, “C-400 Spray Booth” (this NCSE/A is evaluated in Section B.37). Staff found that more specific requirements for sampling (to ensure independence and representativeness) were contained in 400-006, and common-mode scenarios involving sampling errors were discussed; the two NCSE/As together provided an adequate basis for double contingency for these scenarios.

GEN-001 allows the use of the same equipment for both independent measurements, provided that there is a certified source calibration check performed between the two measurements, by a different individual than the individual who performed the previous calibration. Independent measurements are also required to be performed by two different qualified individuals.

The staff also had a concern with the use of process knowledge to characterize waste streams as either potentially fissile or NCS exempt. Condition of Approval (COA) 4.8.1 states:

> “Liquid/solid samples shall be identified and labeled as either potentially fissile or NCS exempt based on NCSA GEN-07, process knowledge, or NCS guidance.
>
> “Note: Process knowledge may be used to label those samples which come from waste streams which have been determined to be less than 1.0 weight percent ^{235}U.”

The staff was concerned with using historical process knowledge to characterize waste streams as PF or NCS exempt. During the on-site licensing review, staff reviewed Engineering Notice EN-C-832-00-002, Rev. 0, which replaced GEN-07 and contained guidance on waste characterization. This reference contained three tables indicating what waste streams were required to be considered as PF, what waste streams were NCS exempt, and what waste streams could or could not be NCS exempt based on process conditions. In the RAI response dated November 22, 2000, USEC stated that process knowledge is only used to characterize waste streams “that are controlled or have previously been determined to be less than 1.0 wt% ^{235}U.” For those waste streams in Table 3 of EN-C-832-00-002 that could be either PF or NCS exempt, the response indicated that “Prior to exempting waste that is listed in Table 3, two people must independently verify...that the material is classified as NCS exempt.” Staff interprets this to mean that “process knowledge” for the purposes of waste characterization means that either the sample must have originated in a facility or process limited to less than 1wt% ^{235}U assay, or from a process verified to have less than 1wt% ^{235}U (by measurement or other controls) prior to pulling the sample. Staff considers this practice to be adequate provided these restrictions are met.

Therefore, the staff finds reasonable assurance of safety in using GEN-001 in conjunction with process-specific NCSE/As to establish dual independent sampling or measurements for NCS.

B.30 NCSA 1493-30, "Handling, Transport (to and from), and Storage of Small Vacuum Pumps from Uranium Containing Systems"

B.30.1 PROCESS DESCRIPTION

NCSE/A 1493-30 covers the handling and storage of small vacuum pumps in Room 144 of Building C-710. These pumps contain lubrication oil that is potentially contaminated with uranium. Several other types of equipment are stored in Room 144, including MD and 1-kg cylinders, but this other equipment must be stored in a dedicated storage location on the floor or in storage cabinets. This room is very small and there are no fissile material processes underway or authorized in this area.

B.30.2 EVALUATION

Each pump is limited to a maximum volume of 1.089 gallons and must be stored with one foot edge-to-edge spacing from each other and from any other fissile or potentially fissile material. Stacking is also prohibited. A 40×40×2 array of vacuum pumps was modeled and demonstrated to be subcritical under both normal and credible abnormal conditions (including interstitial moderation and spacing violations), and assumed a bounding uranium loading of 40wt% uranium in the oil.

The following parameters are controlled for this operation:

1. Enrichment is limited to the plant-wide limit of 5.5wt% ^{235}U assay.
2. Volume is of the pumps' oil reservoir is limited to 1.089 gallons. Small vacuum pumps can only be handled in batches of three pumps.
3. Geometry of the pumps is configuration controlled to specific models considered bounded by the criticality calculations.
4. Concentration of uranium in the oil is limited to 40wt% uranium. The pumps are assumed to seize from the viscosity increase once 40wt% is exceeded.
5. Interaction is controlled administratively by limiting the spacing between the pumps and any potentially fissile/fissile units stored in the room. There are fixed storage locations for MD cylinders stored in trolleys along one way, and the storage cabinets for 1kg cylinders have fixed shelf spacing. In addition, stacking of vacuum pumps is prohibited.

The staff performed confirmatory calculations to verify USEC's calculations. Staff's main concern was with the assumption of 40wt% uranium density in the oil. USEC's calculations demonstrated that a 40×40×2 array would be adequately subcritical assuming 40wt% uranium loading; staff performed confirmatory sensitivity studies showing that a 40×40×2 array would be critical at an optimal uranium loading of ~60wt% uranium. However, the staff recognizes that this is a very conservative model, that would require multiple stacking upsets. The staff also toured this facility and observed that Room 144 was not capable of holding a 40×40 array of small vacuum pumps. Moreover, most of the oil will have been drained from the pumps prior to storing the min Room 144 of C-720.

Staff performed a confirmatory calculation using a 40×40×1 array of small vacuum pumps at the optimal uranium loading of 60wt%, even with the full range of interstitial moderation modeled. USEC also had modeled upsets involving spacing violations with disassembled pumps and 22-liter carboys in the small vacuum pump array, as well as leaking pumps. The maximum quantity of a single leak would be limited to the volume of a single pump, and would spread out on the floor, reducing the system k_{eff}.

Based on these results, although USEC had not established that 40wt% bounds the maximum uranium loading in the pumps, the staff believes that there is an extremely low likelihood of criticality from any credible upsets in the storage of small vacuum pumps in Room 144. Staff did not require revision due to low risk.

B.31 NCSA CAS-004, "C-331 Seal Exhaust/Wet Air Pump Station", 310-002, "C-310 Seal Exhaust/Wet Air Pump Station", CAS-006, "C-333 Seal Exhaust/Wet Air Pump Station", CAS-007, "C-337 Seal Exhaust/Wet Air Pump Station (up to 1.8wt% ^{235}U)", 335-001, "C-335 Seal Exhaust/Wet Air Pump Station (up to 2.0wt% ^{235}U)"

B.31.1 PROCESS DESCRIPTION

The seal exhaust (SX) stations in the C-331 and C-333 buildings are used to provide an exhaust for the seal cavities on the UF6 compressors and purge and evacuation pumps. The seal exhaust headers are maintained at a constant pressure at each cell through the use of control valves on the exhaust header serving each cell. This is accomplished by using two separate lines from the main header, one for the "A" seals and one for the "B" seals. The seal exhaust control valve is located in each of the seal exhaust headers and allows for each of the headers to be regulated to the desired pressure.

The Wet Air (WA) stations in the C-331 and C-333 buildings are used to remove any wet air (a potential source of moderator) from systems. This may be due to being exposed to atmospheric air for the performance of maintenance, a cell placed in standby status, or a cell that has inadvertently experienced wet air inleakage. Systems are also evacuated via a WA station prior to reintroduction of UF_6 in order to minimize contaminants in the enrichment process. The SX and WA stations are combined in the C-310, C-335, and C-337 buildings.

In all SX and WA stations, exhaust gasses are piped through the SX/WA header system, pulled through alumina traps, through the SX or WA pump (as appropriate), and discharged through the vent stack to the atmosphere. The alumina traps are of different sizes and configurations dependent on the enrichment that they will see. Some are in the process of being modified, but modifications will be complete prior to HAUP.

Many of the SX and WA pumps in the higher end of the cascade have their single large internal oil separator (IOS) removed and replaced with two smaller IOSs for criticality reasons. Only the pumps in the C-333 building are exempt from this modification. These higher end pumps are having their oil separators removed prior to HAUP as well as modifying their oil mist eliminators (OME) from one large diameter to two smaller diameter OMEs for criticality reasons.

Some of the SX and WA stations have abandoned in place oil purification systems. These were originally used to filter fresh oil before adding to a pump due to impurities in the oil. Modern oil used at PGDP does not have these impurities and the systems are no longer needed.

NCSE 045 contains the evaluation for the following NCSAs:

! NCSA CAS-004, Rev. 01
! NCSA CAS-006, Rev. 01
! NCSA CAS-007, Rev. 01
! NCSA 310-002, Rev. 02
! NCSA 335-001, Rev. 02

B.31.2 EVALUATION

The principal risk of a criticality in this operation is the buildup of uranium in the oil used to lubricate the pumps. This is due to the unique interaction UF_6 gas has with hydrocarbon oil to produce UF_4 solids.

The following parameters are controlled for this operation:

a. Mass is not controlled directly; however, by ensuring that the Alumina is removed from traps prior to in-place maintenance ensures that the mass of uranium is limited in the event of a trap spill.
b. Enrichment is limited to 5.5 wt% in all buildings except C-333, where it is limited to 1.8 wt% or less.
c. Volume is controlled by limiting the waste collection containers to 5.5 gallons or less.
d. Geometry is controlled through the physical layout of the SX/WA stations and components. Also, slab depth accumulation of contaminated piping and waste materials are limited to 4.75 inches in depth.
e. Concentration/Density is controlled to limit the uranium concentrations that can result in the pump oil reservoir.
f. Interaction is administratively controlled on SX/WA components and waste containers.
g. Neutron Absorption is controlled by relying on the structural materials of the SX/WA components.

One hundred and twenty one different process upsets were identified for this operation. Of these, ninety four scenarios were identified as a potential criticality concern. The remaining twenty seven were deemed to not be a criticality concern. For every potential criticality scenario, a double contingency analysis, contained in Section 4.2 of the NCSE, was performed by the licensee. The licensee determined that all of these scenarios met the double contingency principle given that the required controls given in the NCSE are followed. Staff reviewed both the evaluation and associated supporting documentation.

The oil used in the SX/WA pumps is AQ-NCS controlled to have a density not to exceed 0.87 g/cc. The evaluation assumes a uranium loading of 40wt% based on theoretical predictions and pump SX failure data, and takes a comprehensive look at the possibility of exceeding this loading. Operational data collected over the past decade on pump failure indicated that most of the pumps failed around 8wt% uranium loading, with only one pump reaching 30wt%. Based on the evaluation and technical references reviewed, staff has no reason at this time to believe that 40wt% loading for these pumps is non-conservative. The cases evaluated used this loading for all of the scenarios for conservatism, although in all likelihood, a pump would seize due to viscosity increase before this level of loading could be reached.

Numerous upset conditions were evaluated and shown to be adequately subcritical for all credible scenarios. In all instances, conservative assumptions were made with respect to uranium loading, spacing of components, and structural materials present. The interconnecting piping was ignored in the analysis due to different piping arrangements of each SX/WA station. To bound the piping, reduced spacing between components was used in modeling the system. The addition of the piping to the modeled cases would result in decreased interaction within the system since the pump reservoirs are the drivers, and would only serve to decouple the system.

The controls listed in the evaluation to utilize a passive barrier to preclude the accumulation of uranium-contaminated oil were incorrect. Although the evaluation cites the correct controls, the controls listed are apparently mis-numbered. This has been asked to be corrected prior to HAUP due to the potential to flow down the incorrect controls into the operating procedures. NOTE: This was been corrected as of the last revision, dated December, 2000.

B.32 NCSA GPS-16, “Normetex Pump Maintenance and Testing”

B.32.1 PROCESS DESCRIPTION

The Normetex pumps are used in the product and tails withdrawal facilities at Paducah. GPS-16 covers the maintenance of the Normetex pumps in the C-720 Building. Prior to removal from the process, the pumps are isolated from the cascade and evaucated of UF_6, drained of oil, and the pump openings are covered. Waste generated during pump refurbishing is stored in 5.5-gallon containers. Because of the evacuation step, the pumps are expected to contain only contamination-levels of uranium. It is possible for the pumps to contain deposits of UO_2F_2 through wet air inleakage, however.

B.32.2 EVALUATION

The following parameters are controlled for this operation:

1. Enrichment is limited to the plant-wide limit of 5.5wt% ^{235}U assay.
2. Volume of waste drums will be limited to 5.5 gallons; these drums are configuration controlled. Containers for slugged (*i.e.*, not freely flowing) oil are limited to 2.1 gallons.
3. Interaction is limited administratively to a two foot edge-to-edge spacing bewteen the Normetex pumps and other fissile material. In addition, the dimensions of the pump casing and internals prevent placement of a fissile-bearing container closer than a certain distance to fissile-bearing components in the pump.
4. Although the NCSE/A states that concentration is not controlled, NCSE/A GPS-16 only covers normal pump maintenance. Maintenance involving catastrophic pump failure, that could cause significant loading of uranium in the oil, is not covered by this NCSE/A (greater than safe mass deposits would have to be handled under NCSE/A GEN-10).

The Normetex pumps covered by this NCSE/A are those that have been removed for maintenance without undergoing catastrophic pump failure, and the oil is drained from the pumps before shipping to C-720 for maintenance. (See Section B.26, for a discussion of upsets involving operating pumps). There were only two upsets evaluated that could lead to an accidental criticality: (1) Fissile-bearing container or equipment moved next to the Normetex pump; and (2) uranium loading occurs in the pump oil reservoirs. The first upset would only be a concern if a pump that had undergone a catastrophic failure had failed to have the oil drained prior to moving it the C-720, in addition to the spacing violation. Without catastrophic failure, the pump could contain a deposit of UO_2F_2 from wet air inleakage, but this would be confined to the favorable geometry process gas regions in the pump. The second failure is treated fully in NCSE/A 310-003. In addition, operational data for Normetex pumps at USEC and European facilities was provided in the RAI response. Although pump failures had occurred, there was no operational history of catastrophic pump failure.

This NCSE/A contained several criticality safety limits, although it did not contain any calculations or references to calculation documents. The technical basis for the two-foot spacing limit, the secondary containment pan depth, or the volume limits for slugged and free-flowing oil were not provided. Staff however considered the hazards in this operation bounded by those considered in NCSE/A 310-003, and to be of low risk for reasons stated above. Staff therefore defers its finding of reasonable assurance of safety to the review of NCSE/A 310-003. Staff did not require revision due to low risk.

B.33 NCSA 3973-21, “Field Decontamination”

B.33.1 PROCESS DESCRIPTION

Decontamination operations are undertaken on cascade equipment that is in place in process facilities. Authorized solvents, tools, or vacuum cleaners are used. This NCSE was developed to authorize field decontamination of equipment classified as Uncomplicated

Handling (UH). The NCSE also authorizes field decontamination of equipment classified as Planned Expeditious Handling (PEH) on a case specific basis.
The NCSE also authorizes the cleanup of leaks and spills in process facilities.

This NCSE applies throughout the cascade where field decontamination may be required. The NCSE relies on NCSAs GEN-015 (Waste Handling), GEN-004 (Fissile Vacuum Cleaners), GEN-008 (Sampling), and GEN-010 (PEH Equipment).

B.33.2 EVALUATION

The criticality safety hazard of field decontamination is that fissile material deposits in large cascade equipment may be placed into a critical configuration either due to misidentification or mishandling during or after removal. The two primary accident pathways are failure to correctly identify the deposit type or size and improper removal and handling of fissile material.

The following parameters are controlled for this operation:

1. Enrichment is limited to 5.5 weight% U-235 which is verified prior to commencement of work.
2. Volume of waste containers and vacuum cleaners is limited in accordance with NCSA GEN-015.
3. Geometry of large deposits is controlled by procedural requirements during removal.
4. Moderation is restricted from PEH equipment by covers over openings during decontamination operations and restrictions on use of liquids.
5. Mass is limited during decontamination of PEH equipment by requiring containers and vacuum cleaners to be empty prior to use.

Double contingency for field decontamination is based on multiple controls on deposit identification, limitations on the size and spacing of containers that waste may be placed in, and operation-specific limitations on PEH decontamination.

The staff reviewed the regulatee analysis and notes that the regulatee assumption that fire sprinkler activation during PEH equipment decontamination is unlikely is not supported. The failure to properly establish that fire sprinkler activation is unlikely during PEH decontamination does not affect the result of the overall analysis because PEH decontamination is approved on a case basis and is not considered a doubly contingent operation. PEH decontamination relies on multiple controls on the introduction of moderator such as covers over openings and restrictions on the use of solvents. No argument in this NCSE rests on the unlikeliness of sprinkler activation.

B.34 NCSA 4151-05, “Operation of the Cylinder Weigh Scales”

B.34.1 PROCESS DESCRIPTION

Scales are used in C-333A and C-337A to weigh empty toll cylinders and empty or full product or tails cylinders. The scales consist of a scale cart, a scale pit cover, a scale, and a crane. The scales are located over unsafe geometry concrete pits. The NCSE applies only to two specific cylinder weigh scales in C-333A and C-337A. The regulatee performs feed vaporization activities in these two facilities.

B.34.2 EVALUATION

The criticality safety hazard associated with the cylinder weigh scales is that fissile material may enter the unsafe geometry pits in combination with sufficient moderator to reach a critical configuration due to the size and shape of the pits. This may only happen if fissile material escapes from the cylinder being weighed or is released as a result of activities in the facilities and the pit is concurrently or subsequently filled with moderator.

The following parameters are controlled for this operation:

1. Enrichment is limited to 5.5 weight% U-235.
2. Volume is limited.
3. Moderation is controlled by isolating UF_6 from sources of moderator and restricting the spraying of water onto breached cylinders.

The regulatee has established a reasonable criticality safety limit for the scale pits of 78 pounds of uranium. Double contingency is established through controls on the escape of fissile material, which is unlikely due to cylinder integrity, and procedural controls on handling. Vaporization operations in the facility are not considered by the analysis. The staff considered the arguments surrounding the unlikeliness of a cylinder breach leading to a large fissile material leak to be adequate.

B.35 NCSA 360-005, “C-360 Transfer Station”

B.35.1 PROCESS DESCRIPTION

Fissile material is transferred from a parent cylinder to a receiving, drain or fill cylinder in the C-360 Toll Transfer and Sampling Facility. This evaluation applies to the transfer manifold, piping, valves, scale cart, scale, scale pit, elevator and elevator pit that make up the transfer station in building C-360.

B.35.2 EVALUATION

The criticality safety hazard of

The following parameters are controlled for this operation:
1. Moderation is the primary control for the operation and is provided by controlling the availability of moderating materials.
2. Volume is controlled through limits on container sizes

3. Geometry is controlled through limits on the elevator position and oil pan size.
4. Interaction is controlled through limits on spacing between containers and system components.

Double contingency is established through administrative controls on the amount of pressure the transfer system will be exposed to, administrative controls on system handling, design of the system components to assure structural integrity, a UF_6 detection system to limit total release, limiting oil pans to favorable geometry, volume and spacing control of containers, limits on the availability of moderating material, and administrative requirements to verify cylinder contents prior to transfers. The staff reviewed the regulatee analysis and notes that upsets involving parent cylinders and most moderator intrusion scenarios are covered by other NCSAs.

B.36 NCSA 3973-10-14, "Relocation and Storage of Two 16" NaF Traps Found in the C-400 Receiving Booth"

B.36.1 PROCESS DESCRIPTION

The purpose of this NCSE/A was to provide technical justification for the relocation and storage of two 16" NaF traps found in the C-400 Receiving Booth, and provide the operational limits and conditions necessary to ensure an adequate margin of safety. During the removal of equipment from the C-400 Receiving Booth, two 16" traps were identified as not having nuclear criticality safety approval as potential fissile operations involving greater than 15 g ^{235}U and 1wt% ^{235}U. Based on the results of an NDA, the traps were determined to contain a greater than safe mass uranium content at 2wt% ^{235}U. In order to comply with RCRA requirements, this NCSA was developed.

The traps are nominally 16 inches in diameter with a wall thickness of about 0.25 inches.

B.36.2 EVALUATION

The principal risk of a criticality in this operation is associated with the storage of the traps in proximity to adjacent fissile material activities that as a result of independent process upsets, could challenge the adequacy of the controls identified in this NCSA.

The following parameters are controlled for this operation:

1. Uranium mass is limited by the chemical loading of NaF pellets. For conservatism, the NCSE assumes a UF_6 / NaF loading factor of 2.5. This is considered conservative since operational data show the loading factor not likely to be significantly greater than about 1.

2. Geometry is controlled by the design of the NaF traps. The physical dimensions of the trap provide a geometrically favorable container at 2wt% ^{235}U.

3. Neutron absorption is controlled by requiring the trap walls to have a Monel thickness of at least 0.25 inches.

The margin of criticality safety for this operation is not demonstrable without reliance on the neutron absorption properties of the structural materials. SAR Section 5.2.3.1 states that the NCSE can take credit for the neutron absorption properties of the materials provided an allowance has been made for manufacturing tolerances, dimensional tolerances, corrosion, chemical reactions and uncertainties in the neutron cross sections. Consistent with ANSI/ANS-8.1-1983, Section 4.2.3, concerning nuclear properties relied upon for criticality safety, USEC verifies and maintains a minimum wall thickness to ensure sufficient neutron absorption is available to demonstrate subcriticality. Because the traps are not routinely exposed to acidic environments, there is no credible corrosion mechanism in which the 0.25 inch minimum trap wall thickness will be violated. The NRC concludes, therefore, that verification in non-acidic environments is not required.

During the review, the staff identified a non-conservative NaF bulk density which was utilized in the supporting calculations. Since USEC could demonstrate that modeling NaF at the bulk densities
controlled through the Configuration Management program would not yield k-effectives exceeding the 0.9634 upper safety limit, the NRC concluded that this NCSE discrepancy does not affect the safety of the operation.

By letter dated December 26, 2000, USEC adequately responded to the staff's concerns by committing to either remove the contents of the two traps prior to the next evaluation update or revise the NCSA to incorporate the lower NaF bulk density. Based on the staff's review of the NCSE/A, the staff concludes that the recommended conditions of approval given in the NCSA are appropriate and adequate to assure that the operation meets the double contingency principle. The NCSE is sufficiently documented to demonstrate that all credible upset conditions have been evaluated and controlled to ensure subcriticality. The NCSA adequately documents the conditions of approval based on the results of the NCSE.

B.37 NCSA 400-006, "C-400 Spray Booth"

B.37.1 PROCESS DESCRIPTION

The C-400 Spray Booth is used to clean contaminated equipment such as compressors, converters and valves. The system is also used for cleaning items that have not been previously exposed to process gas. The cleaning methods involve the use of clean plant steam and clean plant water. The Spray Booth drain may be used for disposal of aqueous uranium-bearing solutions that are compatible with Spray Booth components and operations.

Prior to placement in the Spray Booth, the uranium content and enrichment of items to be cleaned is determined. Maximum allowed enrichment is 5.5wt% ^{235}U and the total uranium mass must be less than the uncomplicated handling (UH) mass limit corresponding to the enrichment of the most highly enriched item.

The cleaning operation involves cleaning a component with clean water using the high pressure sprayer or plant water wands or steam using a steam lance. The resulting solution drains from the equipment item onto the Spray Booth floor and into the floor drain. The solution accumulates in the drain and floor until the return pump is used to transfer the solution from the Spray Booth drain to the Spray Booth storage tanks. The solution is accumulated and stored in the storage tanks until it is transferred to the C-400 or C-409 receiving tanks.

Aqueous fissile solutions ($\leq$ 5.5wt% ^{235}U) may be disposed of in the Spray Booth drain. Such solutions comply with chemical compatibility criteria to protect the integrity of the Spray Booth components and to protect uranium recovery/precipitation operations. The solutions are brought to the Spray Booth in either AQ-NCS approved 5.5 gallon waste drums or AQ-NCS approved 21-liter polybottles. On a single fissile solution container is allowed in the Spray Booth at a time.

Solutions in the Spray Booth storage tanks with enrichments less than 1wt% ^{235}U may be transferred to the C-400 receiving tanks. Solutions in the Spray Booth storage tanks with enrichment equal to or greater than 1wt% ^{235}U can be diluted by mixing with depleted uranium solutions before transfer to the C-400 receiving tanks. Solutions with enrichment equal to or greater than 1wt% ^{235}U, but less than 1.5wt% ^{235}U and with uranium concentration less than or equal to 100 g/liter, may be transferred without dilution to the C-400 receiving tanks. Solutions in the Spray Booth storage tanks with enrichment no more than 5.5wt% ^{235}U may be transferred to the C-409 receiving tanks. Depleted uranium solutions used to dilute the high enrichment (equal to or greater than 1wt% ^{235}U) solutions may be generated in the Spray Booth during cleaning of items with depleted uranium or may be solutions generated in other C-400 operations, such as cylinder wash.

B.37.2 EVALUATION

The principal risk of criticality in this operation is associated with the accumulation of enriched uranium (*i.e.*, uranium enriched up to 5.5wt% ^{235}U) solution in unfavorable geometry. For solution transfers to the C-400 Uranium Recovery system, or to the C-409 Uranium Precipitation process, uranium enrichment and/or uranium concentrations are restricted per NCSAs 3973-09 and 409-001, respectively.

The following parameters are controlled in this operation:

a. Geometry is controlled to ensure that safe slab height is not exceeded for the Spray Booth floor and the Spray Booth roof. Geometry is controlled in the design of the Spray Booth floor drain, storage tanks, and waste containers.

b. Neutron absorption is relied on in this evaluation in that a tank wall thickness of 0.25 inch of stainless steel is credited. Walls of fissile waste containers are also credited.

In the December 6, 2000, request for additional information, the staff identified two technical concerns that required NCSE revisions. These concerns were:

1. NCSE scenario 17 dike location, NCSE scenario 28 filtrate tank location and NCSE scenario 30 presence of the dividing wall were not identified as AQ-NCS; and

2. Potentially singly contingent vulnerabilities existed in the plant water and plant air interfaces to the Spray Booth. In particular, the NCSE credited the unlikelihood of either a plant water or plant air system failure such that siphoning from the Spray Booth storage tanks was possible. Although the NCSA identified a requirement to verify plant water or plant air pressure before the plant water or plant air systems are used in the operation, the NCSE did not provide an adequate basis for the availability or reliability of these systems during the time period the systems were actually used.

By letter dated December 26, 2000, USEC adequately responded to these concerns. In its response, USEC committed to identifying the SSCs identified above as AQ-NCS. In addition, USEC committed to continuously monitor the plant water and plant air pressures during the short time that those systems are being used to add water or inject air into the uranium solution system. Because the plant air system is used following solution transfer, and the solution transferred is limited to an always safe concentration, the concentration controls from the transfer operation to the C-409 Uranium Precipitation process provide an adequate second leg of double contingency regarding plant air backflow. Based on the staff's review of the revised NCSE/A, the staff concludes that the recommended conditions of approval given in the NCSA are appropriate and adequate to assure that the operation meets the double contingency principle. The NCSE is sufficiently documented to demonstrate that all credible upset conditions have been evaluated and controlled to ensure subcriticality. The NCSA adequately documents the conditions of approval based on the results of the NCSE.

B.38 NCSA GEN-13, "Operation and Maintenance of the High Efficiency Particulate Airborne (HEPA) System in Buildings C-310 and C-360"

B.38.1 PROCESS DESCRIPTION

GEN-13 covers the operation and maintenance of High Efficiency Particulate Air (HEPA) filters at various placed throughout the Paducah plant-site. Although HEPA filters are used extensively at the site, there are many areas where the potential for build-up of fissile material in the filters is minimal or not credible. The HEPA systems evaluated consist of the inlet hood, ductwork, filter units, and exhaust fans, primarily in the C-310 and C-360

Buildings. Filters removed from these areas are handled under GEN-13; filters removed from other plant areas are specifically excluded (*i.e.*, do not have an active NCSA for fissile operations). The filters consist of a high-efficiency fiberglass medium for trapping airborne particulates.

B.38.2 EVALUATION

The main criticality hazard in this operation is the build-up of an unsafe amount of uranium in unfavorable geometry filter housings, either from an acute release or long-term build-up.

The following parameters are controlled for this operation:

1. Mass of uranium in the filter housings is controlled by periodic visual inspections and NDA measurements, the nature of the operations connected to the ventilation system, and the UF_6 Detection System.
2. Moderation is limited administratively by requiring a visual inspection or NDA measurement within 7 days after the introduction of a large amount of water into the ventilation system.
3. Enrichment is limited to the plant-wide limit of 5.5wt% ^{235}U assay.
4. Interaction is limited administratively by requiring a two-foot edge-to-edge spacing between fissile material and the filter enclosures.

Visual inspections or NDA measurements of the unfavorable geometry regions of the HEPA ventilation system are conducted every two years, to preclude the long-term buildup of unsafe deposits of uranium. In its December 26, 2000, RAI response, USEC provided data demonstrating that even with the maximum number of batch operations per year–assuming as a source term the C-310 Product Withdrawal, C-360 Sampling Station, and C-360 Autoclaves whose filters are included in this evaluation–it would take many years before an unsafe mass could accumulate. This evaluation was made assuming that each time one of these batch operations is conducted, it results in a one-gram release; the number of batch operations is also at least a factor of 1.5 times more than the typically number of operations in a given time period. Releases greater than one-gram would presumably result in a visible cloud, and a visual inspection/NDA is required within one week following a visible release. Even with these conservative assumptions, it would require a minimum of 9.3 years (Letter GDP-00-0226 states 4.6 years, which is a typographical error). Even if one inspection period is missed, an unsafe mass cannot accumulate.

For acute releases in the C-360 Sampling Station, the maximum acute release would be the contents of one 2S-cylinder (4.9 lb UF_6). For the C-310 Withdrawal Station, the maximum release amount is bounded by the amount that would go undetected by the UF_6 Detection System. GEN-13 assumes that the filters would clog before 5.1 lb uranium could accumulate on the filters. In its RAI response of December 26, 2000, USEC provided the basis for this measurement, which assumes a 15-micron layer of UO_2F_2 particulate over the entire surface area of the filter medium, which is 50 times the pore size. Even when combined with the maximum release amount from the C-360 and C-310 operations, this would be significantly less than the minimum critical mass for these operations. In addition, USEC performed calculations of an infinite filter array. These calculations demonstrated that even with more than 5 times the amount of mass assumed in the 5.1-lb calculation (82-microns), the system would remain subcritical. In addition, another study showed that even with full flooding, the HEPA filter (at 15-microns) would remain subcritical. The staff recognizes that this model is extremely conservative because it assumes an infinite array of filter medium.

Staff also questioned the basis for allowing one week before responding to an intrusion of liquid water into the filter housing. USEC did not justify the basis for the one-week period, but instead acknowledged that it was somewhat arbitrary. However, multiple failures would be required before an unsafe mass (~22.3 kgU) could accumulate in the filter. The sensitivity studies above demonstrate that the filters are subcritical even fully flooded. Based on these studies, the time required to accumulate an unsafe mass in the filters, and the controls established in GEN-13, the staff finds the risk of criticality to be acceptably low.

B.39 NCSA 409-001, “C-409 Uranium Precipitation”

B.39.1 PROCESS DESCRIPTION

Uranium solutions having an enrichment greater than 1.5wt% ^{235}U are transferred to the C-409 Uranium Recovery system. The system includes a solution acidifying and storage tank system, a precipitation tank system, a rotary vacuum precipitation system, and a filtrate tank system. The solution acidifying and storage tanks include four stainless steel solution acidifying tanks, 12 stainless steel solution storage tanks, two solution transfer pumps, a nitric acid day tank, and a nitric acid metering tank. The precipitation system consists of six stainless steel precipitation tanks, a slurry transfer pump, and a NaOH feed drum. The rotary vacuum precipitation system consists of a rotary drum filter, a vacuum receiver, a vacuum pump and a filtrate transfer pump. The filtrate tank system consists of eight stainless steel filtrate tanks, a filtrate transfer pump and a 1200 gallon portable tank. A stainless steel floor pan with an approximate depth of 1 inch covers the floor area encompassing the Uranium Recovery system.

The solution acidifying and storage tanks are nominal 10 inch diameter. The tanks are used in groups of four connected by equalizing lines and each group is considered one tank set. One tank set is used to acidify the uranium solution, the remaining three tank sets are used to store the acidified solution. All storage, acidifying, precipitation and filtrate tanks share a common overflow and vent line. If a tank should overflow, the overflow is discharged onto the stainless steel floor pan. The vent line also exhausts near the floor pan.

The precipitation system tanks are nominal 10 inch diameter. Each of these tanks is equipped with two liquid sensors. Should the liquid level reach these switches, the solution storage transfer pump, the slurry transfer pump and the filtrate pump will trip off and a visual and audible alarm will sound. An upper level switch also actuates a visual and audible alarm.

Concentration/density is used as a control for transfers of filtrate to the 1200 gallon portable tank. For this activity, concentration is limited to 0.1 g U/liter.

B.39.2 EVALUATION

The principal risk of a criticality is associated with accumulations of enriched uranium (i.e., uranium enriched to greater than 1wt%) in unfavorable geometry.

The following parameters are controlled for this operation:

1. Geometry is controlled to ensure that safe slab heigh is not exceeded for the floor pan. Geometry is also controlled in the design of the storage tanks, system piping and waste containers.

2. Concentration is controlled in the transfer of fissile solution from the precipitation tank-set to the rotary drum filter. Concentration is also controlled in the transfer of filtrate solution to the filtrate disposal tank. The concentration is limited to 0.1 g U/liter.

3. Neutron absorption is relied on in this evaluation for crediting the 0.25 inch thick stainless steel wall for the solution tanks. Walls of the fissile waste containers were also credited.

In the December 6, 2000, request for additional information, the staff identified one technical concern that required NCSE revisions. This concern involved using an 11 inch thick void space between the bottom of the horizontal header and the top of the 1.75 inch floor pan in calculational models. Since the calculations could be negatively impacted (i.e., k-effective increases) by model changes that reduce this void space, the staff concluded that the 11 inch thick void space should be declared as a criticality safety related item.

By letter dated December 26, 2000, USEC adequately responded to this concern. In its response, USEC committed to identifying the SSC identified above as AQ-NCS. Based on the staff’s review of the NCSE/A, the staff concludes that the recommended conditions of approval given in the NCSA are appropriate and adequate to assure that the operation meets the double contingency principle. The NCSE is sufficiently documented to demonstrate that all credible upset conditions have been evaluated and controlled to ensure subcriticality. The NCSA adequately documents the conditions of approval based on the results of the NCSE.

B.40 NCSA CAS-014, “Cascade Holding Drums”

NCSA CAS-014 was originally included in the list of evaluations to be reviewed for the HAUP amendment request, but could not be located at the time of its submittal. Because of time constraints and because the staff believed it had enough information to make a programmatic assessment with the remaining NCSE/As, this was dropped from the review.

B.41 NCSA CAS-005, “Cascade Surge Drums”

B.41.1 PROCESS DESCRIPTION

Each of the ‘00' and ‘000' cascade buildings contain several banks of surge drums which are used for various operations, including:

a. Storage of dry air which has been used to purge UF_6 systems;
b. Evacuating cells and piping that contain UF_6;
c. Storage of atmospheric air and R-144 resulting from inleakage;
d. Storage of essentially pure gaseous UF_6;
e. Emergency vacuum source for the seal exhaust system; and
f. Storage of treatment gasses.

The surge drums are each 8 feet in diameter and nominally 40 feet long with a volume of approximately 2000 cubic feet. Groups of drums, which are called banks, are tied together through common headers. Banks include anywhere from 3 to 5 drums each. Drums within a bank cannot be physically isolated from each other, and each bank header has a surge drum block valve that connects it to the building evacuation header. The banks of surge drums are enclosed within insulated rooms that are heated to maintain the temperature high enough to prevent condensation or desublimation of the process gas.

Since this operation involves fissile material, an NCSE was performed by USEC to demonstrate that this operation meets the double contingency principle when the recommended conditions of approval given in NCSA CAS-005 are met.

1. EVALUATION

The principal risk of a criticality in the operation of the surge drums is desublimation or condensation of UF_6 and condensation of HF. Condensation of HF is of concern since it is the primary source of moderation present within the surge drums.

The following parameters are controlled for this operation:

a. Mass is not controlled directly, however, limits are in place to maintain UF_6 in a gaseous phase to assure that the mass of UF_6 in the surge drums does not compromise the structural integrity of the drums.
b. Enrichment is limited to 5.5 weight percent ^{235}U.
c. Volume is controlled by limiting waste collection containers, vacuums, and batches of pressure transmitters to 5.5 gallons or less.
d. Geometry is not controlled, however it is fixed by the surge drum dimensions.
e. Moderation is controlled by maintaining the temperature and pressure ranges under which the surge drums operate to prevent desublimation or condensation of UF_6 and condensation of HF.
f. Interaction is only controlled such as it interacts with other NCSAs, and is handled under those specific NCSAs.

For every potential criticality scenario, a double contingency analysis, contained in Section 4.1 of the NCSE, was performed by the licensee. The licensee determined that all of these scenarios met the double contingency principle given that the required controls given in the NCSE are followed. Staff reviewed both the evaluation and associated supporting documentation and considers the recommended conditions of approval given in the NCSE to be appropriate and adequate to assure that this operation meets the double contingency principle and that this document demonstrates an adequate safety basis for operation of the surge drums.

B.41.2 EVALUATION

The principal risk of a criticality in the operation of the surge drums is desublimation or condensation of UF_6 and condensation of HF. Condensation of HF is of concern since it is the primary source of moderation present within the surge drums.

The following parameters are controlled for this operation:

a. Mass is not controlled directly, however, limits are in place to maintain UF_6 in a gaseous phase to assure that the mass of UF_6 in the surge drums does not compromise the structural integrity of the drums.
b. Enrichment is limited to 5.5 weight percent ^{235}U.
c. Volume is controlled by limiting waste collection containers, vacuums, and batches of pressure transmitters to 5.5 gallons or less.
d. Geometry is not controlled, however it is fixed by the surge drum dimensions.
e. Moderation is controlled by maintaining the temperature and pressure ranges under which the surge drums operate to prevent desublimation or condensation of UF_6 and condensation of HF.
f. Interaction is only controlled such as it interacts with other NCSAs, and is handled under those specific NCSAs.

Thirty-seven different process upsets were identified for this operation, all of which indicated a potential criticality concern. For every potential criticality scenario, a double contingency analysis, contained in Section 4.1 of the NCSE, was performed by the licensee. The licensee determined that all of these scenarios met the double contingency principle given that the required controls given in the NCSE are followed. Staff reviewed both the evaluation and associated supporting documentation and considers the recommended conditions of approval given in the NCSE to be appropriate and adequate to assure that this operation meets the double contingency principle and that this document demonstrates an adequate safety basis for operation of the surge drums.

B.42 NCSA 310-006, “C-310 Cylinder Burp Station”

B.42.1 PROCESS DESCRIPTION

The C-310 Cylinder Burp Station is designed to vent light gases (including R-114, ClF_3, F_2, and HF) within solid UF_6 product cylinders for product purity. Liquid UF_6 from the enrichment cascade contains various light gases that are fed into cylinders along with the product at product withdrawal. The solidification of UF_6 results in expansion, which decreases the void space in the cylinders and causes these gases to become pressurized. The light gases are fed through a pigtail into the North and South Bank NaF Traps in C-310, which are then vented to the atmosphere through the C-310 stack. NCSE/A covers only the operation of the Cylinder Burp Station itself, not the operation and maintenance of product withdrawal or the North and South Bank NaF Traps.

The Cylinder Burp Station consists of three cylinder stations, pigtails and associated piping, the Number 1 and 2 space recorders and associated instrumentation, and the NaF Traps and UF_6 evacuation header. The Burp Station itself is located under an overhang on the outside of the C-310 Building.

B.42.2 EVALUATION

The main criticality hazards in the Cylinder Burp Station are from potential moderation of the 14-ton solid UF_6 cylinders. Potential sources of moderator are limited because the cylinders, unlike those in the UF_6 cylinder yard (covered by GEN-03), are protected by an overhanging roof. There is, however, a wand that is provided for cylinder cooling. Movement of liquid UF_6 cylinders into the Burp Station and burping of liquid UF_6 cylinders is prohibited.

The following parameters are controlled for this operation:

1. Mass control is claimed for this operation; it is asserted but not demonstrated that greater than 13.9kg U will not escape from a breached solid UF_6 cylinder. The staff did not consider cylinder burping to be under mass control because this was not demonstrated and because of the large mass of UF_6 in a 14-ton cylinder.
2. Enrichment is limited to the plant-wide limit of 5.5wt% ^{235}U assay. However, the North Bank NaF Traps are limited to 2.0wt% ^{235}U and are physically disconnected if the cylinder to be burped exceeds 2wt% ^{235}U assay.
3. Geometry of the maximum slab depth that could occur due to accumulation of liquids in the Cylinder Burp Station is limited by a dike to 3.5 inches. The Space Recorders are favorable geometry.
4. Moderation is controlled by relying on the structural integrity of the UF_6 cylinders, a remote cutoff to the spray water supply in the event of a cylinder breach, and the overhanging roof.
5. Interaction between potentially fissile/fissile containers and UF_6 cylinders, space recorders, and NaF traps, is administratively limited to a two-foot edge-to-edge spacing.

With regard to neutron absorption, NCSE-048 states: "Not controlled. Neutron absorbing material is assumed not present in this analysis. Full credit may be taken for materials of construction." The staff reviewed the criticality safety calculations, which take credit for the presence of the carbon steel cylinder shell. Although technically taking credit for neutron absorption, the staff did not pursue this issue in this case because the cylinders are required to meet certain standards (such as ANSI-N14.1) before the y can be used for transportation, that requires a robust construction. Compliance with these standards should provide adequate assurance that the thickness and material properties are conservatively bounded by the models.

The main criticality hazard is the introduction of water from the cooling water spray wand. USEC maintained that less than 7kg of water is needed to achieve criticality in a 14-ton cylinder, but did not demonstrate the maximum amount of water that could be introduced through a breach. Following the staff's review of the NCSE/A, USEC provided calculations demonstrating that a product cylinder would be doubly contingent for scenarios involving a large breach in the cylinder yards. This finding was based on a model using conservative assumptions about the maximum breach size, maximum expected rainfall, and a conservative configuration of the uranium-water mixture. These calculations were done in support of the staff's review of NCSE/A GEN-03, and found to provide an adequate basis for doubly contingency.

The conditions following a breach in the Cylinder Burp Station would be bounded by those evaluated in GEN-03, with the exception that the flow rate of the cooling water spray is not controlled and would be concentrated in a much smaller area. Staff toured the Burp Station and observed the equipment used for the cooling water spray. The wand used consists of a long pipe connected to a flexible hose with many perforations for cooling water along its length, and is held over the top of the cylinder during cooling. There are administrative requirements to inspect cylinders for breaches prior to burping, disconnect the water spray in the event of a cylinder breach, and remove the spray wand and valve out the spray water for five minutes after the cylinder valve is opened. The emergency cutoff valve for the cooling spray water is remotely accessible to operators and is considered AQ-NCS.

The requirement to isolate the cooling spray water for five minutes following the beginning of cylinder burping is based on historically observed explosions due reactions involving light gases in the cylinders. Some of these light gases are explosion hazards and tend to react with contaminants in the Burp Station piping/equipment. These explosions have all been observed within seconds following the initiation of burping; operators stated that none of these explosions have resulted in a breach to the cylinder. This is the most likely cause of a breach following initiation of burping. Handling requirements to prevent a breach during cylinder movement are also in place.

Staff raised another concern with the release of material from the product cylinders. USEC asserted that the maximum amount of material that could be released from a breached solid cylinder was 13.9 kgU. In the event of a solid UF_6 cylinder breach, liquid water is required as a motive force for moving uranium to the storm sewer. The 3.5-inch dike prevents an unsafe depth of pre-existing water fromaccumulating in the vicinity of the Burp Station. Moreover, cylinder burping is not allowed then there is measurable water or snow reaching the cylinder or when snow is mounded up higher than the dike curb. Staff observed during system walk-downs that there is a grate flush with the bottom of the concrete path where water would be diverted rather than overflowing the top of the dike. There are several feet of space between the nearest cylinder and the grate; staff believes that a considerable flow rate would be needed to provide the motive force for washing uranium into the storm sewer.

Double contingency for breaches involving solid UF_6 cylinders is based on: (1) cylinder integrity; and (2) favorable geometry configuration in the event of a cylinder breach. This latter barrier includes measures to prevent significant moderation, since liquid water is required to transport the material to an unfavorable geometry collection point. Double contingency for liquid cylinders is based on: (1) cylinder integrity; and (2) the prohibition on movement and burping of liquid UF_6 cylinders.

The staff noted that there are many administrative actions, including cylinder handling requirements and housekeeping practices in the Cylinder Burp Station that are credited for double contingency. Some of these are considered administrative controls, but many–including cylinder handling requirements–are listed as assumptions rather than controls. This could be a concern if administrative assumptions relied on in the NCSE/A have not been programmatically controlled. This issue is discussed in the programmatic section on the use of “unlikely events” in lieu of controls, in CER Section 4.1.

Staff concluded, however, that based on the results of the review of GEN-03, the cylinder burp operation did not introduce any hazards that were not bounded by the GEN-03 evaluation, which the staff found to be acceptable, with the following exceptions: (1) presence of the cooling water spray; and (2) the explosion hazard leading to a cylinder breach. Administrative controls on moderation to prevent criticality in the event of a cylinder breach were considered adequate as analyzed in this NCSE/A.

B.43 NCSA CAS-002, “Operation and Maintenance of the UF_6 Cascade”

B.43.1 PROCESS DESCRIPTION

The nature of the gaseous diffusion process used at PGDP utilizes large, unfavorable geometry process equipment, which requires specific operational controls to reduce the potential for an inadvertent nuclear criticality. This evaluation strays from the previous evaluation in that it addresses only the routine operation of the enrichment cascade (the shutdown cascade was evaluated in NCSE/A CAS-011; Section B.45). Normal operation refers to the aspects of cascade operation involving normal processing with the motors running and with UF_6 being pumped from cell to cell through the cascade. It also includes cells running off-stream with some UF_6 inventory and cells shutdown, but remaining on-stream with a UF_6 flow. All other shutdown operations are covered separately under CAS-011 and its accompanying evaluation, NCSE 039.

The cascade itself consists of approximately 1800 stages located in four buildings, with each stage comprised of a motor, compressor, converter, gas cooler, control valve, and associated piping. The cascade is connected in series and spans all of the buildings using inter-building tie lines. Gas can flow up and down the enrichment cascade.

During the compression of UF_6 gas, the large amounts of heat generated are cooled by the process coolant system consisting of a refrigerant loop and a water loop. Because of the danger of the water moderator entering into the cascade, the refrigerant loop is used and controlled as a barrier between the process gas and the water loop (RCW). The RCW system normally only has trace quantities of uranium present and is not considered a fissile material operation.

B.43.2 EVALUATION

The principal risk of criticality in the gaseous diffusion cascade is moderation. Although there is no specific limit on moderation entry into the cascade, the inherent chemical behavior of the UF_6 present in the cascade provides mitigation of moderation entry by keeping the hydrogenous portion of the reaction products in a gaseous state (typically in the form of HF). Numerous controls seek to minimize moisture and organic material inleakage into the cascade, and as such, moderation control is maintained by maintaining the cascade integrity.

Therefore, the major threats to criticality safety involve maintaining the integrity of the cascade. The integrity can be challenged by hot metal reactions (HMRs), wet air inleakage, exothermic reactions, and other moderator entry through penetrations (*e.g*., oil, RCW, fire water, etc.).

In the original USEC submission to the NRC, a substantial effort was made to demonstrate double contingency for all identified hazards. Unfortunately, in evaluating the HMR and fire water entry scenarios, credit was taken for the maximum penetration size in the cascade, which in effect would limit the amount of moderator entry under emergency conditions (*i.e*., fire suppression system activated). NCR staff attempted to confirm this maximum penetration size through USEC staff interviews, calculations, and the available historical data on incidents that resulted in penetration of the cascade, however, no sound basis for this maximum penetration could be identified. In response to the ensuing Request for Additional Information (RAI), USEC acknowledged this lack of basis and reinstated the operation of the cascade as a singly contingent operation.

For all credible wet air inleakage events with the cascade operating no pathway leading to a criticality was identified due to the continuous presence of a fluorinating environment. This environment reacts immediately with water vapor to form HF and dry UO_2F_2, which prevents the moisture from excessively moderating any existing deposit, as well as serving to seal any small penetrations due to the formation of a new deposit.

The following parameters are controlled for this operation:

a. Enrichment is limited to 5.5 wt% in all buildings except C-333, where it is limited to 1.8 wt% or less.
b. Moderation is controlled by maintaining a leak tight cascade; however, some small inleakage is expected. This inleakage of moderator typically reacts

with UF_6 or other treatment chemicals present in the cascade, limiting the amount of free water. Significant leaks require shutdown of the affected cells.

c. Concentration/Density is controlled such that UF_6 remains in a gaseous state to the extent possible. Gaseous UF_6 does not have sufficient density to go critical at the enrichment limit.

d. Interaction is administratively controlled by limiting the proximity to the cascade to which potentially fissile containers and equipment may approach, typically 2 feet edge-to-edge for UH equipment.

For every potential criticality scenario, a double contingency analysis was performed by the licensee. The licensee determined that all of these scenarios met the double contingency principle given that the required controls given in the NCSE are followed with the exception of the HMR scenario and fire water entry into the cascade. Staff reviewed both the evaluation and associated supporting documentation. Numerous upset conditions were evaluated and shown to be adequately subcritical for all credible scenarios, and in all instances, conservative assumptions were made.

B.44 NCSA 3974-09, “South Bank NaF Traps”

B.44.1 PROCESS DESCRIPTION

The NaF traps are used to remove trace quantities of UF_6 from evacuated gases originating in Paducah product cylinders. The trapping system consists of a linear array of seven traps having an approximate 7-foot center-to-center separation.

During the absorption cycle (cylinder burping), the gas stream is routed from the cylinder at the burp station through the traps to the atmosphere. The gas enters the bottom of the trap and exits the top of the trap. Loading of the NaF trap is determined by daily stack emission measurements and routine gamma scans performed by operations personnel. When necessary, the NaF pellets are regenerated by heating the traps to 700 degrees Fahrenheit and evacuating the UF_6 to the C-310 cascade.

The traps are constructed of 10 inch schedule 20 Monel. The outside diameter for schedule 20 pipe is 10.75 inches with a wall thickness of 0.250 inches. A mill tolerance of 12.5% is used to reduce the wall thickness further to bound expected manufacturing variations. Two of the constituents used in the calculational models, sodium and copper, are not addressed in the validation. NCSE Section 4.2.9 demonstrates subcriticality (k-eff less 0.9634) given the substitution of these materials with fluorine and hydrogen, which are addressed in the validation.

B.44.2 EVALUATION

The principle risk of a criticality in this operation is associated with the accumulation of enriched uranium (i.e., uranium enriched to greater than 1wt% ^{235}U) in unfavorable geometry.

The following parameters are controlled for this operation:

a. Uranium mass is limited by the chemical loading of NaF pellets. For conservatism, the NCSE assumes a UF_6 / NaF loading factor of 2.5. This is considered conservative since operational data show the loading factor not likely to be significantly greater than about 1.

b. Geometry is controlled by the design of the NaF traps. The physical dimensions of the trap provide a geometrically favorable container at 5.5wt% ^{235}U.

c. Neutron absorption is controlled by requiring the trap walls to have a Monel thickness of at least 0.213 inches.

The margin of criticality safety for this operation is not demonstrable without reliance on the neutron absorption properties of the structural materials. SAR Section 5.2.3.1 states that the NCSE can take credit for the neutron absorption properties of the materials provided an allowance has been made for manufacturing tolerances, dimensional tolerances, corrosion, chemical reactions and uncertainties in the neutron cross sections. Consistent with ANSI/ANS-8.1-1983, Section 4.2.3, concerning nuclear properties relied upon for criticality safety, USEC verifies and maintains a minimum wall thickness to ensure sufficient neutron absorption is available to demonstrate subcriticality. Because the traps are not routinely exposed to acidic environments, there is no credible corrosion mechanism in which the 0.213 inch minimum trap wall thickness will be violated. The NRC concludes, therefore, that verification in non-acidic environments is not required.

Based on the staff's review of the NCSE/A, the staff concludes that the recommended conditions of approval given in the NCSA are appropriate and adequate to assure that the operation meets the double contingency principle. The NCSE is sufficiently documented to demonstrate that all credible upset conditions have been evaluated and controlled to ensure subcriticality. The NCSA adequately documents the conditions of approval based on the results of the NCSE.

B.45 NCSA CAS-011, "Shutdown of Cascade with & without Inventory"

B.45.1 PROCESS DESCRIPTION

Like its operational counterpart CAS-002 (Section B.43), the nature of the gaseous diffusion process used at PGDP utilizes large, unfavorable geometry process equipment, which requires specific operational controls to reduce the potential for an inadvertent nuclear criticality. This evaluation strays from its predecessors in the fact that it addresses only shutdown operations of the enrichment cascade. It covers the equipment and processes associated with the operating cascade during and after shutdown, as well as during restart. It also applies to running cells off-stream with a coolant negative or UF_6 negative (including less than 0.5 psia of UF_6 inventory); stopping the motors with or without UF_6 inventory; cell treatment operations; inleakage of RCW into the cascade. Before shutdown and after restart operations are covered separately under CAS-002 and its accompanying evaluation, NCSE 052.

B.45.2 EVALUATION

The principal risk of criticality in a shutdown portion of the cascade is moderation. Since there are no direct pathways for moderation to enter shutdown equipment (due to maintaining the system integrity) under normal circumstances, only an abnormal or emergency situation could challenge this integrity. For most analyzed situations, double contingency can be shown to provide an adequate margin of safety to prevent a criticality from occurring. However, in the instance of a fire in close proximity to the cascade, the operation can only be shown to be singly contingent. Unlike in NCSE/A GEN-10, the fire is the initiating event to a breach, whereas for cascade equipment removal there is a pre-existing breach caused by the removal activities. Thus, CAS-011 is singly contingent on cascade integrity in the case of a large fire in close proximity to the cascade.

Although the analyzed fire scenarios are singly contingent, the measures in place appear substantial enough to provide reasonable assurance of safety. The rarity of a major fire in close proximity to he cascade (when normal operations are underway) coupled with controls in the fire safety program and the training of emergency responders are together adequate to provide reasonable assurance of maintaining the cascade integrity.

Wet air inleakage into equipment is also of concern since shutdown equipment doesn't have the same fluorinating environment as an operating cascade to provide a limiting force against moderator intrusion. Based on calculations performed by USEC, it would take over 15 years of wet air inleakage from normal atmospheric conditions to moderate a deposit to the point where a criticality would be possible. The control to dry shutdown equipment at least once every 10 years by heating up the equipment is sufficient to prevent moderation of a deposit through wet air inleakage.

Other moderator entry into shutdown equipment is limited by monitoring the maximum penetration size and external water exposure.

The following parameters are controlled for this operation:

1. Enrichment is limited to 5.5 wt% in all buildings except C-333, where it is limited to 1.8 wt% or less.
2. Moderation is controlled by maintaining a leak tight cascade; however, some small inleakage is expected. This inleakage of moderator typically reacts with residual UF_6 or other treatment chemicals that are still present in the cascade immediately after shutdown. By requiring heating up the shutdown equipment every decade, moderation control is maintained.
3. Concentration/Density is not controlled, however, deposits are assumed to be UO_2F_2 due to its greater reactivity compared to UF_6 and UF_4.

4. Interaction is administratively controlled by limiting the proximity to the cascade to which potentially fissile containers and equipment may approach, typically 2 feet edge-to-edge for UH equipment.

For every potential criticality scenario, a double contingency analysis was performed by the licensee. The licensee determined that all of these scenarios met the double contingency principle given that the required controls given in the NCSE are followed with the exception of the fire water entry into the cascade. Staff reviewed both the evaluation and associated supporting documentation. USEC had recognized the need for revising this NCSE/A prior to the conduct of the licensing review for operational reasons, and the staff required that this be performed prior to approval of HAUP. Numerous upset conditions were evaluated and shown to be adequately subcritical for all credible scenarios, and in all instances, conservative assumptions were made.

B.46 NCSA 3974-05, “Product Withdrawal System”

B.46.1 PROCESS DESCRIPTION

The product withdrawal system is located in Buildings C-310 and C-310A. The system provides three pathways to allow for the simultaneous withdrawal of two product streams with different ^{235}U enrichments. Typically, two of the pathways will be utilized as a redundant path for product withdrawal, with one pathway idle. Any of the pathways may be used to withdraw material of any assay up to the plant maximum.

The product material that is withdrawn is in response to customer orders, and may range up to approximately 5.0 wt% ^{235}U. However, a nominal increase in assay (approximately 0.25 wt% ^{235}U) may occur in the C-310 low speed cells integral to the product withdrawal system to ensure the correct product assay in the withdrawal cylinder, therefore, the maximum allowed at PGDP was set at 5.5 wt% to bound this nominal increase and to add a small cushion for withdrawals.

The withdrawal system is composed of 3 Normetex pumps, 3 condensers, 2 accumulators, and 2 withdrawal stations. UF_6 gas from the cascade is compressed by the Normetex pumps and sent on to the condensers to liquify the gas by cooling it to 147-165 ^{o}F. The accumulators are used as temporary liquid UF_6 storage whenever the UF_6 condensing rate exceeds the cylinder filling rate or when a cylinder is changed out.

The withdrawal stations use an air powered cart to move cylinder up to withdrawal positions, where they are connected with piping known as a pigtail (because of the curly shape). When placed in a withdrawal position, the cylinder on the cart rest on a scale that measures the amount of UF_6 being withdrawn. The scales are contained in a pit beneath the withdrawal positions, and each is equipped with liquid sensors to detect any accumulation of water withing the scale pits.

After filling a cylinder with UF_6, the cylinder is moved to the cylinder yard where it is left to cool either by normal convection cooling. The cooling can be supplemented with water spraying when burping the cylinder. Water cooling is covered under the cylinder burp evaluation, NCSA 310-006.

Since this operation involves fissile material, an NCSE was performed by USEC to demonstrate that this operation meets the double contingency principle when the recommended conditions of approval given in NCSA 310-004 are met.

B.46.2 EVALUATION

The risks of a criticality in the operation of product withdrawal is broken into four main areas: Normetex pumps and associated piping; condensers and the cooling system; product withdrawal stations and associated piping (including the scales and carts); and the accumulators and their associated piping, including rupture disks. The primary concern for criticality in the product withdrawal system is moderation control, and therefore engineered and administrative controls are used to provide protection against moderator.

The following parameters are controlled for this operation:

1. Mass is not controlled during normal operations since this is a continuous flow operation, but in accident conditions, both automatic and operator actions mitigate or terminate any release of UF_6.
2. Enrichment is limited to 5.5 weight percent ^{235}U.
3. Volume is not controlled for the withdrawal system, however volume limitations are placed on unsafe volume components associated with the process, including containers and the sup pit.
4. Geometry is a controlled parameter for the condensers, and is used in the supporting calculations.
5. Moderation is controlled in the sump pits, condensers, and the UF_6 cylinders
6. Interaction is controlled administratively for accumulations of used pigtails and spacing of groups of small open containers.

For all scenarios with potential criticality concerns, a double contingency analysis, contained in Section 4.1 of the NCSE, was performed by USEC. USEC determined that all of these scenarios met the double contingency principle given that the required controls given in the NCSE are followed. Staff reviewed both the evaluation and associated supporting documentation and considers the recommended conditions of approval given in the NCSE to be appropriate and adequate to assure that this operation meets the double contingency principle and that this document demonstrates an adequate safety basis for operation of the product withdrawal system.

B.47 NCSA 37A-001, "C-337A Feed Station Relief Drums"

B.47.1 PROCESS DESCRIPTION

The C-337A Feed Station relief drums consist of two large"Type O" UF_6 cylinders on the outside of the C-337A building. Type O cylinders are not approved for transportation and storage of enriched UF_6, but the drums were specifically evaluation in NCSE/A 37A-001 for NCS. The purpose of these drums is to purge the contents of UF_6 feed cylinders to a vacuum source, to provide pressure relief for the steam autoclaves, line purging, or hot or cold burping of cylinders. UF_6 feed from the steam autoclaves provides the source of material for the relief drums, which consists primarily of burped light gases, fluorinating agents, steam, or other low uranium-concentration materials.

The drums differ from cascade surge drums primarily because they are outside the building and subject to precipitation, vehicle accidents, and other environmental hazards. There are two drums connected to the autoclaves through a feed header, are steam heated and wrapped with insulation to maintain the temperature within the drums, and are equipped with a condensate drain.

B.47.2 EVALUATION

The following parameters are controlled for this operation:

1. Mass is not controlled for this operation directly because the drums will contain uranium regularly and may have deposits due to wet air inleakage. However, under normal conditions the material being fed to the drums is expected to have low uranium concentration.
2. Moderation is controlled by maintaining the relief drum pressure $\leq$ 5 psia and keeping the temperature above 140°F. These process controls prevent the condensation of HF (the interior of the drums is expected to be a fluorinating environment) and the hydration of any deposits present. The masses of water, carbon, and HF are limited as established by NCSE-037.
3. Spacing is maintained between the drums by the fixed design, and the spacing between the drums and other potentially fissile/fissile containers is administratively controlled.

Note: Previous operation of the relief drums relied primarily on mass control; mass control would limit the operation to less than 40lb uranium at 5.5wt% ^{235}U assay.

The main criticality concern with the relief drums is moderation. There are several potential sources of moderation in the drums, including: (1) water, cascade coolant, hydrocarbons, and HF condensation; (2) moderation from the steam supply; and (3) precipitation following a drum breach. These three sources of moderation are discussed below:

The only source of moderator that could be introduced through the feed header to the relief drums would be from non-condensible gases in the UF_6 cylinders. There is a requirement to perform a cold pressure check prior to cylinder feeding to detect this.

Hydration of deposits within the drums is prevented by maintaining the relief drum pressure $\leq$ 5 psia and the temperature above 140°F. The drum pressure is monitored daily. The reviewed walked-down both the relief drums and the UF_6 feed process in the C-337 Building, and noted that the autoclaves are equipped with two rupture disks in series on the header to the relief drums. Since the drums provide over-pressure protection for the autoclaves (evaluated in NCSE-037), a pressure transient could cause loss of moderation control. The rupture disks as equipped with AQ-NCS actuation alarms to alert operators. Beyond the rupture disks, there is an AQ-NCS Autoclave High Pressure Isolation System to provide isolation of the relief drums within 15 seconds of detecting a pre-determined high autoclave pressure. In addition, past operational experience indicates that a pressure transient would only result in a momentary increase in the pressure, due to the large surge volume.

There was a concern when the NCSE was written about the possible presence of historical liquid water in the relief drums. Although the drums are typically subjected to a fluorinating environment, and have been for several years, maintenance of a fluorinating environment had not been controlled. In addition, NDA measurement indicated the presence of deposits (expected to form from the reaction of UF_6 with water). The only water that should be present in the drums should be that physically bound within the deposits. However, the NCSE conservatively assumes a residual water inventory in each drum of 2.11 lb (whereas the safe limit is 10 kg according to NCSE-037). Although the staff did not verify these calculations in NCSE-037 directly, the low pressure and highly fluorinating environment provide reasonable assurance that no significant unbound water exists in the relief drums. Thus, the only credible means of moderating the deposits sufficiently to cause criticality would result from the addition of liquid water.

Moderation from the steam supply is prevented by the design of the relief drums. The heated steam piping is wrapped around the outside of the relief drum, and surrounded with a layer of insulation several inches thick. Before it is possible to get steam into the drum's interior, it would be necessary to breach both the steam line and the thick shell of the drum. USEC has acknowledged that a steam leak could cause the shell to become corroded; however, due to the robust nature of the shell (that of a Type O UF_6 cylinder), this is expected to take a considerable length of time to occur. Staff considers USEC's assertion that there would be noticeable degradation in the insulation surrounding the steam piping before the cylinder could corrode all the way through to be reasonable. In addition, the pressure drop in steam would be noticeable to the operators, and a breach in the cylinder would prevent a pressure of 5 psia from being maintained in the drums. For all these reasons, the staff does not consider moderation due to a steam line breach to be a credible event.

Because the drums are outside the building, it is possible for precipitation to get into the drums in the event of a massive rupture. The evaluation done to demonstrate double contingency in GEN-03 (for 14-ton UF_6 cylinders) does not absolutely bound this type of cylinder, which has not been evaluated for 5.5wt% ^{235}U material. Two factors that mitigate the risk, however, are that the moderation intrusion calculations were very conservative, and that the relief drums are not expected to contain mass quantities of uranium under normal conditions. They are expected to contain low-density gas with only contamination levels of uranium under normal conditions. Although there could be UO_2F_2 deposits caused by wet air inleakage, the mass of uranium is likely to be significantly smaller than that encountered in the 14-ton cylinders that are evaluated in GEN-03. Therefore, the system should be adequately subcritical for this scenario.

There are engineered features designed to protect the cylinders against a breach, or limit the size of the breach. The drums rest on cradles that are not welded or bolted to the drums, and the feed piping is limited to a diameter of 2.5-inches. USEC maintained that upon vehicle impact, the drum would be knocked out of its cradle and only the weakest structural member, the feed header, would rupture. This would limit the maximum size of a breach to 2.5-inches in diameter. The staff observed several vehicles passing by the road next to the relief drums, including a straddle carrier holding a 14-ton cylinder. There is also a plant air tank of similar dimensions to the drums between the two drums, that would tend to absorb some of the impact (although this is not credited in the evaluation). There are also concrete pillars and the wall of the C-337 Building surrounding these drums, which will restrict the range of motion from a catastrophic transportation accident. Staff does not believe that USEC's evaluation adequately demonstrates that such an accident cannot result in a breach larger than 2.5-inches across, but concludes that the risk of criticality resulting from such a breach is relatively low based on the 14-ton cylinder calculations done in support of GEN-03.

Based on these controls, the staff has reasonable assurance that there is double contingency protection against a criticality caused by liquid moderator intrusion.

B.48 NCSA 3971-07, "Operation and Maintenance of the C-310 Tops Purge Trapping System"

B.48.1 PROCESS DESCRIPTION

The top ten cells in the cascade are used for the removal of light molecular weight gases that are first passed through an alumina trapping system to remove residual UF_6 before being exhausted to the atmosphere.

The alumina traps are configured in two banks consisting of 13 traps each. The traps are constructed of 10 inch schedule 20 pipe which is conservatively assumed to be entirely carbon steel. The outside diameter for schedule 20 pipe is 10.75 inches with a wall thickness of 0.250 inches. A mill tolerance of 12.5% is used to reduce the wall thickness further to bound expected manufacturing variations.

B.48.2 EVALUATION

The principal risk of criticality in this operation is associated with the accumulation of enriched uranium (i.e., uranium enriched to greater than 1wt% ^{235}U) in unfavorable geometry.

The following parameters are controlled for this operation:

1. Uranium mass is limited by the chemical loading of alumina pellets. The NCSE requires that the bulk density of alumina be controlled to not fall below 0.7 g/cm^3.

2. Geometry is controlled by the design of the alumina traps. The physical dimensions of the trap provide a geometrically favorable container at 5.5wt% ^{235}U.

3. Neutron absorption is controlled by requiring the trap walls to have a thickness of at least 0.213 inches.

The margin of criticality safety for this operation is not demonstrable without reliance on the neutron absorption properties of the structural materials. SAR Section 5.2.3.1 states that the NCSE can take credit for the neutron absorption properties of the materials provided an allowance has been made for manufacturing tolerances, dimensional tolerances, corrosion, chemical reactions and uncertainties in the neutron cross sections. Consistent with ANSI/ANS-8.1-1983, Section 4.2.3, concerning nuclear properties relied upon for criticality safety, USEC verifies and maintains a minimum wall thickness to ensure sufficient neutron absorption is available to demonstrate subcriticality. Because the traps are not routinely exposed to acidic environments, there is no credible corrosion mechanism in which the 0.213 inch minimum trap wall thickness will be violated. The NRC concludes, therefore, that verification in non-acidic environments is not required.

In the December 6, 2000, request for additional information, the staff identified one technical concern that required NCSE revisions. This concern involved calculational models using alumina with a bulk density greater than the bulk density maintained through the configuration management program. Because USEC could demonstrate that using the lower bulk density would not yield k-effectives exceeding the 0.9634 upper safety limit, the staff concluded that this concern does not affect the safety of the operation.

By letter dated December 26, 2000, USEC committed to addressing this concern during the next evaluation update. Based on this review and USEC's commitment, the staff concludes that this NCSE/A provides reasonable assurance of safety, and meets the double contingency principle.

B.49 NCSA 710-005, "C-710 Drain and C-712 Neutralization Pit"

B.49.1 PROCESS DESCRIPTION

There are several pathways for fissile solution entry into the drain system. Operations that involve the routine introduction of uranium bearing material into the drain system include the auto tube washing system, cylinder wash system, plugged tube removal and straight tube cleaning. Pathways that could inadvertently introduce fissile solution into the drain system include cup and sink drains.

The drain system ties into a collection pit located southwest of the building, the C-712 pit. The pit's primary function is to collect acidic solutions. The pit is lined with acid brick and has a central baffle for diverting incoming flow. The pit overflows into the sanitary sewer system.

B.49.2 EVALUATION

The principal risk of criticality in this operation is associated with accumulation of enriched uranium (i.e., uranium enriched to greater than 1wt% ^{235}U) solution in unfavorable geometry.

The following parameters are controlled for this operation:

a. Uranium mass is limited by the fixed volumes of individual sample containers. The worst case uranium transfer to the drain system is bounded by the 22.3 kg uranium subcritical limit.

b. Concentration of solutions entering the drain system is controlled such that the worst case uranium transfer to the drain system will not exceed the 123 g U/liter subcritical limit.

The C-712 Acid Pit NCSE/A was originally submitted to NRC in support of USEC's request for excluding that facility from the criticality accident alarm system requirements of 10 CFR 76.89. During that review, the NRC staff noted that two NCSEs supporting Laboratory operations which affect the C-712 Acid Pit lacked sufficient detail to ascertain the adequacy of the administrative controls and the assumptions and nature of the process to determine whether an immediate safety concern existed in the Acid Pit. The staff found that the C-712 Acid Pit NCSE credited mass control for double contingency but did not ensure that the mass limits evaluated were consistent with the Laboratory NCSEs. As a result, the staff was unable to demonstrate mass control for the Acid Pit based on the information contained in the C-712 Acid Pit NCSE. The staff was also unable to evaluate the C-712 Acid Pit NCSE's assertion that the dynamic conditions of the Acid Pit preclude precipitation of uranium without further discussions with Laboratory personnel. Since factors which affect the ability to precipitate, such as pH, ratio of water to uranium, potential for introducing caustic agents into the drain system, and sampling methodology, were not addressed in the C-712 Acid Pit NCSE, the staff was unable to demonstrate the geometry and concentration controls as relied upon by USEC for the Acid Pit based on the information contained in the NCSE.

NRC asked USEC on March 24, and March 30, 2000, for quantitative uranium data that showed how much uranium is actually in both the acid pit and the storm sewer system located down stream of the acid pit. USEC could not provide the staff with quantitative amounts of uranium in either the acid pit or storm sewer. Instead, USEC discussed average sample data accumulated over the last 3 years to demonstrate that negligible quantities of uranium actually are drained to the pit. Since such average data may not be representative of actual concentrations in either the pit or the storm sewer, NRC questioned USEC on the utility of this information in determining whether a safety question exists.

In the acid pit NCSE, the staff found that USEC assumed that more than 156 kg U could be transferred to the acid pit as a result of credible upset conditions in the C-710 lab. During the two telephone conferences, USEC stated that it relies upon specific laboratory NCSAs to control the amount of uranium entering the pit so that environmental discharge limits in the storm sewer system are not exceeded. USEC also stated that relying on the controls described in these NCSAs (as opposed to the controls described in the acid pit NCSA), results in only 43 kg U ending up in the acid pit (vice 96 kg). Because only 32 kg U at 5.5wt% ^{235}U is necessary to sustain a criticality in the storm sewer, NRC asked USEC to demonstrate why the existing controls relied upon in the specific laboratory NCSAs provide adequate assurance that a criticality is not possible in the storm sewer. USEC committed to calling the staff on March 31, 2000 to provide its safety appraisal of the situation. Based on that telephone conference, USEC voluntarily implemented a 2 weight percent (wt%) enrichment limit on activities affecting the Acid Pit until technical differences with the NRC staff could be resolved. NRC concluded that this interim control eliminated the immediate concern regarding a criticality.

The NRC staff arrived at the PGDP site on April 11, 2000, to determine whether an immediate safety concern existed at the authorized enrichment limit of 5.5wt% ^{235}U. Based on the staff's on-site review of the activities associated with the operation of the acid pit, the staff determined that the maximum credible mass of uranium which could be introduced into the acid pit exceeds the minimum critical mass for 5.5wt% ^{235}U or 32 kg U. Although this amount is the minimum necessary for criticality under spherical and fully reflected conditions, the staff determined that the actual conditions of the pit in precluding formation of spherical deposits allowed for a critical uranium loading significantly larger than the maximum credible mass. The staff, therefore, concluded that no immediate safety concern exists at the currently authorized 5.5wt% ^{235}U and that the operation of the acid pit as actually conducted can be shown to meet the double contingency principle. However, the description of the acid pit operation in the Acid Pit NCSE (and supported by two Laboratory NCSEs) did not accurately describe the actual configuration of the plant. Moreover, the Acid Pit NCSE failed to establish double contingency protection over this operation, and therefore required revision.

During the HAUP review, the NRC reviewed the revised NCSE/A and associated documentation and determined that (1) the hazards that were identified were appropriate and (2) the administrative controls, engineered controls, maintenance activities, and surveillance activities that were identified are sufficient for the hazards identified. Overall, the recommended conditions of approval given in the NCSE are appropriate and adequate to ensure that these operations will not result in a criticality.

The staff determined that USEC had established adequate controls to demonstrate compliance with the double contingency principle, for all credible upset scenarios.

Preliminary

Appendix A
Specific Examples of Programmatic Issues Identified During Review

As a result of the staff's review of the Paducah HAUP CAR, the NCSE/As selected for detailed review, review of other documents (RAI responses, technical references, *etc*.) described in this report, and direct observation of plant operations and interactions with plant personnel, several over-arching programmatic deficiencies were identified. These are described in the bullets below. Many of the NCSE/As mentioned exhibited multiple programmatic concerns, and some examples cut across multiple areas because they are so closely interrelated.

Some of the most safety-significant examples (resulting either in actual changes to plant operations or to the underlying safety documentation) follows, cross-referenced to the affected NCSE/A:

! USEC exhibited an over-reliance on administrative controls, instead of following the preferred design approach. No basis was given for using administrative over engineered controls for NCS.

   1. In GEN-10, USEC opted to use the purely administrative control of dual visual inspections over the use of NDA in making mass determinations (Section B.8). *This resulted in an operations change from visual to NDA controls.*

   2. In 310-003, USEC opted to use monthly sampling of the oil instead of the fluoride cell or gamma monitor recommended in its own technical reports, to monitor the build-up of uranium in oil (Section B.26). *This resulted in a certificate condition that required use of an active-engineered control and increased sampling frequency.*

! Older NCSE/As did not adequately document the basis for double contingency, including some assumptions and items relied on for NCS. Not all items were configuration controlled.

   1. In 710-005, USEC had taken credit for C-710 Laboratory conditions and limits, as well as flow conditions in the C-712 Acid Pit; these were not controlled in the NCSE/A (Section B.49). *This resulted in a revised NCSE/A to clarify the safety basis and the addition of a control to monitor flow in the Acid Pit.*

   2. In 310-003, several items relied on for criticality safety had been omitted from the list of Safety Related Items, including the pump scroll tolerances, purge gas piping diameter, the pump oil cap, and the pump isolation valves

(Section B.26). *This resulted in the addition of several new AQ-NCS controls.*

3. In GPS-19, double contingency was only established for the handling of UH deposits; accident scenarios involving the loss of spacing and moderation control for PEH deposits was singly contingent. However, USEC did not differentiate between these in the NCSE/A (Section B.17). *This resulted in a revised NCSE/A, and new TSR requirements for PEH equipment handling to ensure moderation and spacing control.*

4. In 400-006, several items relied on for criticality safety had been omitted from the list of Safety Related Items, including location of a dike and filtrate tank, and a dividing wall (Section B.37). *This resulted in a revised NCSE/A and the addition of several new AQ-NCS controls.*

5. In 409-001, a gap credited for criticality safety had been omitted from the list of Safety Related Items (Section B.39). *This resulted in the addition of a new AQ-NCS control.*

6. NCSE-045 (Section B.31) had the wrong controls for an accident scenario involving the accumulation of uranium-contaminated oil; this could have resulted in the wrong controls being flowed down into procedures. *This resulted in a revised NCSE/A and revised list of controls.*

! Assumptions relied on in NCSE/As were not documented, did not have a technical basis, or did not have the underlying conditions upon which the assumption was based controlled.

1. In CAS-002, double contingency was based on an assumed maximum breach size in the cascade equipment (Section B.43). USEC did not request removal of the TSR controls for the cascade (based on change in singly contingent status), so there were not controls removed. USEC did not demonstrate the conservatism of this assumption. *This resulted in a revised NCSE/A and a reversion from double to single contingency status.*

2. In 310-003, USEC did not demonstrate assumptions about the uranium particle distribution in oil or the maximum leak rate under anticipated normal conditions (Section B.26). *This is an open issue on this preliminary CER and resolution may require a certification condition.*

3. In GEN-10, assumptions about why concrete present around PEH equipment was not modeled were not justified in the NCSE/A (Section B.8). *This resulted in performance of new calculations to demonstrate that it was conservative to exclude the concrete.*

4. In 360-006, there was an implicit assumption in that the calculations performed were done at 5.0wt% ^{235}U assay, although this NCSE/A was intended to bound plant operations at 5.5wt% ^{235}U assay (Section B.3). This assumption was non-conservative. *As a result, the system was cut-and-capped, and abandoned in place.*

5. In GEN-03, assumptions about the maximum breach size of large UF_6 cylinders was not demonstrated, leading staff to question the double contingent status of these cylinders. *This resulted in performance of new calculations to demonstrate double contingency.*

! Some singly contingent scenarios were identified that did not have TSRs or were not identified as singly contingent in the NCSE/As (*e.g.*, equipment removal activities, PEH spacing, UF_6 cascade operations).

1. In GEN-10, GEN-10-01, and GPS-25, singly contingent accident scenarios involving the intrusion of sprinkler water or spacing violations into PEH-bearing equipment were not identified as such in the NCSE/As (Sections B.8, B.10, and B.18). No TSR limits had been established as required for singly contingent processes by TSR 3.11.5. In GPS-19, some accident sequences were doubly contingent only for UH deposits but were singly contingent for PEH deposits, although the NCSE discussed both cases as being doubly contingent (Section B.17). *This resulted in revised NCSE/As and the establishment of new TSR controls.*

2. In CAS-002, double contigency was based on an assumed maximum breach size in the cascade equipment (Section B.43). Following discussions with NRC staff, USEC acknowledged that occurrence of a cascade breach resulting from a hot metal reaction was a singly contingent scenario. *This resulted in a revised NCSE/A.*

3. In CAS-011, USEC had failed to identify a singly contingent scenario involving the occurrence of fires close to shutdown cascade equipment (Section B.45). *This resulted in a revised NCSE/A that required crediting Fire Safety Program measures.*

! There is no periodic review of plant NCSE/As to ensure they bound as-built plant conditions.

1. In GPS-01, the Compressor Disassembly Pit was analyzed assuming no credible sources of moderator within the Pit (Section B.12). A hydraulic line

was subsequently discovered in the Pit that invalidated this assumption. *This resulted in the NCSE/A being withdrawn.*

2. In 710-005, the description of the Acid Pit did not accurately reflect actual operations in the C-710 Laboratory or C-712 Neutralization Pit (Section B.49). *This resulted in a revised NCSE/A and the addition of a control to monitor flow in the Acid Pit.*

3. In 360-006, USEC originally determined that this NCSE/A bounded the operation of C-360 Cold Trapping System at 5.5wt% ^{235}U assay, even though 360-006 was only done at 5.0wt% assay (Section B.3). *As a result, the system was cut-and-capped, and abandoned in place.*

! Selection of controls and determination of unlikelihood is subjective and based on engineering judgement with a technical basis. There is no mechanism for determining control reliability or taking plant performance data into consideration.

1. In GEN-10, USEC relied on activation of the sprinkler system during PEH equipment removal (which involves acetylene torches) being an unlikely event (Section B.8), but could not demonstrate that this was sufficiently unlikely. USEC also did not demonstrate the adequacy of dual visual inspections in determining the mass of uranium deposits. *This resulted in a revised NCSE/A and several new TSR controls.*

2. In 310-003, USEC relied on the occurrence of a leak larger than 10 cm^3/min being unlikely (Section B.26), but could not demonstrate this. USEC also did not demonstrate the bounding uranium density assumed. *This resulted in a commitment form USEC to revise the words in the SAR to address this issue.*

3. The use of unlikely events without requirements to control the conditions relied upon in making the event unlikely was identified as a programmatic weakness during the SAR review (Section 4.1).

! Construction materials credited as neutron absorbers were not required to be controlled and their compositions were not required to be verified.

1. In 310-006, the materials of construction were modeled but neutron absorption was not recognized as being credited for NCS (Section B.42). In the following NCSE/As, neutron absorbers were explicitly credited for NCS: NCSE-045 (Section B.31), 400-006 (Section B.37), 3973-10-14 (Section B.36), 409-001 (Section B.39), and 3971-07 (Section B.48). This list should

be sufficient to demonstrate that neutron absorbers are used extensively in modeling process equipment. Although no NCSE/As were required to be revised and no plant changes were made in connection with this, verification of construction materials credited for NCS was identified as a programmatic weakness during the SAR review (Section 4.2.1).